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Generation IV Roadmap

R&D Scope Report for Nonclassical Reactor Systems

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ABSTRACT

The primary focus of the Nonclassical Concepts Technical Working Group (TWG-4) is to evaluate novel nuclear energy systems with design features that put them in a category clearly different from water, gas, or liquid metal cooled reactors. Novel reactor concepts are evaluated based on their potential to improve one or more of the sustainability, safety, and economic goals of Generation IV Nuclear Energy Systems. The working group has gathered a total of 32 concepts; 28 of them meet the Generation IV requirement of fission-based self-sustained criticality, or are related to nuclear energy systems that meet this requirement. These concepts feature a range of reactor designs with nuclear fuel in gaseous solid and liquid phases, with no coolant or with nonconventional coolant, and with a variety of energy conversion systems. Following the executive summary are two brief sections, the first introduces the nonclassical reactor concept, and the second describes the evaluation methodology. The primary focus of this report, however, is on the two sections that describe respectively the Nonclassical Nuclear Energy Systems and the Crosscut Design issues, which together cover 26 of the viable concepts. The first of these provides the in-depth discussion and evaluation of the six primary concept sets, the second discusses four crosscut design issues that are generically applicable to more than one reactor concept. A fifth section discusses the other two of the 28 viable concepts, however, these two were submitted late and were not evaluated in the screening for potential to meet the Generation IV goals. Appendix C discusses the remaining four concepts that fall within the Nonclassical category but not necessarily within Generation IV guidelines. The references made to sustainability, safety, and economies of concepts are taken directly from input received by the TWG-4. Based on their characterizing design features, nonclassical reactor concepts are divided into five concept sets that include: Liquid Core Reactors, Gas Core Reactors, Nonconventional Cooled Reactors, Non-Convection Cooled Reactors, and Direct Energy Conversion Reactors. A comparative evaluation of these concept sets identified the first three with the highest potential to meet or exceed Generation IV reactor performance goals. Non-Convection Cooled and Direct Energy Conversion Reactors, which are primarily designed with specific applications in mind, were each evaluated and will not be considered for any further Generation IV development.

EXECUTIVE SUMMARY

The primary focus of the Nonclassical Concepts Technical Working Group (TWG-4) is to evaluate novel nuclear energy systems with design features that put them in a category clearly different from water, gas, or liquid metal cooled reactors. Novel nuclear energy system concepts are evaluated based on their potential to improve one or more of the sustainability, safety, and economic goals of Generation IV Nuclear Energy Systems. The working group has gathered a total of 32 concepts, 28 of them meet the Generation IV goal of fission based self sustained criticality. Twenty-two of the concepts are encompassed in the N1 to N17 Department of Energy code names for submissions; six concepts were submitted but not coded, and four concepts from the Russian liquid metal-cooled reactor group at the Obninsk Institute are included as late submissions even though they properly belong to TWG-3. These concepts feature a range of reactor designs with nuclear fuel in gaseous and liquid phases, with no coolant or with nonconventional coolant, and with a variety of energy conversion systems.

Based on the primary design features of the nonclassical nuclear energy system concepts submitted, five "concept sets" are defined as follows:

1. Liquid Core Reactors
2. Gas Core Reactors
3. Nonconventional Coolant Reactors
4. Non-Convection Cooled Reactors
5. Direct Energy Conversion Reactors.

Nuclear Energy System Concepts

Liquid Core Reactor Concept

The Liquid Core Reactor Concept set includes two subsets: molten salt fuel and eutectic metallic fuel reactors.

The molten salt reactors are usually fueled with uranium or thorium fluorides dissolved in a mixture of molten lithium and beryllium fluorides (other salts based on Na or Zr fluoride could be used). This molten salt core combines the functions of fuel and coolant into one. The flowing molten salt from the core directly transports energy to an intermediate heat exchanger. Molten fluoride salts are chemically stable at high temperatures and have a very low vapor pressure. Because the molten salt reactor operates at near atmospheric pressures, a heavy primary vessel for containment is not needed. The high temperature (>700°C) of the nuclear heat source allows for high efficiency power conversion as well as the potential for producing other energy products such as hydrogen. The technical feasibility of molten salt reactors was demonstrated through the performance of a number of large-scale experiments in the 1960s.

Eutectic metallic fuel reactors use a mixture of uranium or plutonium and a low neutron absorbing metal. A separate heat transport system is needed to remove energy from the core. The eutectic metallic fuel concept is in a very early stage of development.

Gas Core Reactor Concepts

Gas core reactor concepts comprise all reactors that are fueled by stable uranium compounds in gaseous or vapor phase at operating temperatures that are usually above 1,500 K. A condensable or non-condensable working fluid is added to enhance the heat transport property of the flowing core material. Gas core reactors feature power generation and conversion at very high temperatures. The heat is used in stages. A closed magnetohydrodynamic (MHD) power generation cycle is first used to directly process and convert fission power at temperatures of 1,800–2,500 K. The rejected heat from the MHD power cycle is used to power single or multiple gas turbine and/or superheated steam cycles. The overall efficiency of the Gas Core Reactor systems is in excess of 60%.

Gas core and molten salt reactors feature minimized waste and fully integrated fuel cycles in which fission products are, in principle, continuously separated from the fuel. High neutron flux and geometrical flexibility of gas core reactors allow for both the design of highly efficient converter fuel cycles, and for minor actinide burning.

Nonconventional Coolant Reactors

Nonconventional coolant reactors use molten salts or high boiling point organic liquids to remove core power at low pressure and provide heat at high temperatures. A particular concept, the Advanced High Temperature Molten-Salt-Cooled Reactor (AHTR), is designed around a graphite-matrix-fuel cooled with a molten fluoride salt ($2LiF\text{-}BeF_2$). The molten salt coolant leaves the core at temperatures above 1,000°C to produce hydrogen from water or to generate electricity using an indirect power cycle. Organic coolant compounds feature a high concentration of hydrogen and a low vapor pressure. They are intended to replace the moderating power and enhance the heat transport capability of water.

Non-Convection Coolant Reactors

Non-convection cooled reactor concepts include all reactor designs that do not use bulk convective cooling to transport the core power. Reactors in this concept set are designed to conductively or radiatively transfer heat either to a passive heat transport medium (heatpipe), or to a static in-core energy conversion system such as thermionic diodes, thermoelectric converters, etc. Reactors in this concept set are characterized by relatively high operation temperature and relatively higher conversion efficiency due to the use of an in-core topping cycle.

Direct Energy Conversion Systems

Direct energy conversion (DEC) systems are built from very thin uranium dioxide foil fuel. Fission fragments carry most of their energy and electric charge out of the microns-thick foils to direct power conversion devices. There are a multiplicity of magnetic field configurations and high voltage electric fields that can be used to convert the kinetic energy of the charged fission fragments to electric current. The energy conversion ratio for these reactors is a strong function of the thinness of the fuel foil. The requirement to have very thin fuel foils may make it difficult for this design to achieve criticality or operate to any reasonable level of fuel burn-up. DEC concepts also require highly enriched fuel.

Summary Appraisal

A comparative evaluation of these concept sets identified Gas and Liquid Core reactors, as well as the Advanced High Temperature Molten-Salt-Cooled reactor systems, as having the highest potential to meet or exceed Generation IV reactor performance goals. Non-Convection Cooled and Direct Energy Conversion Reactors, which are primarily designed with specific applications in mind, and

Organic-Cooled Reactors that may result in limited evolutionary improvement compared with similar commercially available systems, were each evaluated and eliminated from any further consideration.

Table E-1 collects all the averaged scores that the TWG-4 arrived at for potential to meet or exceed Generation IV goals, tabulated according to the broadest practical groupings of nonclassical reactor concepts submitted. In the Evaluation Methodology Group scoring terms, a score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems. Compared to light water reactors, most of the nonclassical reactor concepts are at earlier stages of development. Several of the concepts were initially unfamiliar to the members of the nonclassical technical working group and were therefore harder to accurately evaluate. For each concept set, an "Advocate" subgroup was assigned to gather information and prepare presentations with sufficient details on design characteristics including sustainability, safety, and economy features. Subsequent to the presentation by the concept advocate that was followed by a group discussion, a polling was conducted to score the concept set in each subgoal. The scoring range was 1–10, with a score of 5 in mind for light water reactors. Although under each goal there were a number of criteria and metrics, all of which were scored, only the category averages are presented in the report. The evaluation tables also include the spread of opinion among members.

Many of the concepts discussed require a potentially much larger "stretch" of the technology compared to classical water, gas, or liquid metal cooled reactor systems. With the liquid core systems and

Table E-1. Summary table of averaged evaluation scores for nonclassical concept groups.

Generation IV Goals	LCR[a]	GCR[b]	AHTR[c]	OCR[d]	NCCR[e]	DEC[f]
Sustainability 1 Fuel Utilization & Impact	++	++	+	+	–	–
Sustainability 2 Waste Management	++	++	+	=	–	–
Sustainability 3 Weapons Proliferation Resistance	=	+	=	=	–	–
Safety and Reliability 1 Worker Safety and Plant Reliability	+	+	+	+	+	=
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	+	+	+	=	+	=
Safety and Reliability 3 Mitigation against Offsite Emergency	+	+	+	=	+	=
Economics 1 Life-Cycle Cost Advantage	+	+	+	=	–	– –
Economics 2 Low Financial Risk	+	=	+	=	–	– –

a. Liquid Core Reactors
b. Gas Core Reactors
c. Advanced High-T (Molten-Salt Cooled) Reactor
d. Organic Cooled Reactors
e. Non-Convection Cooled Reactors
f. Direct Energy Conversion Reactors

the gas/vapor core, very significant advances can be made in the areas of resource utilization and waste minimization with the in situ burning of actinides. For the gas/vapor core reactors, especially dramatic improvements can be made in conversion efficiencies, and, as a result one would hope, in economics. These systems have excellent nonproliferation characteristics. Due to one to two orders of magnitude lower fuel inventory and continuous separation of fission fragments, gas/vapor core reactors could potentially eliminate the need for Offsite Emergency Planning, which is a most challenging safety goal for the Generation IV reactors.

If we look especially at the gas/vapor core reactors, in order to make such substantial leaps, there is also substantial technical challenge and risk. Ultimately such a concept may not prove feasible, at least at the impressive performance level that is presented by concept advocates. However, it is believed that such risk is still consistent with the Generation IV goals, given the payoff implicit in core exit temperatures of 2,500 K to 3,000 K. Thirty years should be sufficient to determine if the gas/vapor core reactor concept is viable, and bring it to the point where it is ready for commercialization. The data uncertainty and the competitive weighing of high potential versus technical risk for gas/vapor core reactors are well reflected in the relatively large spread of opinion among TWG-4 members.

On the other end of the spectrum are evolutionary concepts, such as the Organic Cooled Reactor, that feature more moderate performance at lower technology risk. Due to many commonalities and overlaps with existing reactor systems, evolutionary concepts are at higher levels of technology maturity. The lower data uncertainty in evolutionary concepts produces a lower spread of opinion as it is reflected in the evaluation of the Organic Cooled Reactor compared to less evolved concepts.

The concept sets that passed the initial screening for potential are:

- Molten Salt Core Reactors
- Gas Core Reactors
- Advanced High Temperature Molten Salt (Nonconventional) Cooled Reactor
- Organic Cooled Reactors (also Nonconventional-Cooled).

The concepts that did not pass the screening for potential were,

- Solid-State Heatpipe Cooled Reactors (Non-Convection Cooled)
- Direct Energy Conversion Reactors.

The Crosscut Issues

The technical breadth of the Nonclassical Concepts presented to the TWG-4 has been substantial. The TWG evaluated additional concepts with specific crosscut design features and forwarded them for further evaluation to special groups designated to work on crosscut issues.

The crosscut design issues are comprised of the following list:

1. Minimum Waste Fuel Cycles
2. Diverse Energy Product Reactors

3. Modular Deployable Reactor Systems

4. Advanced Fuel Materials

5. Alternative Power Conversion Cycles.

Minimum Waste Fuel Cycles

Waste minimization is aimed at improving the environmental impact by reduction of the probability and minimum value of the long-term dose, by optimizing the fuel and thermodynamic energy conversion cycles as well as the use of radioisotopes. Molten salt and gaseous fuel reactors feature fully integrated fuel cycles including online separation of fission products. Over the next several decades the worldwide demand for radioisotopes will grow dramatically. The growth will be fueled by significantly expanded use of the radioisotopes in medicine and food irradiation. Additional demand will come from uses in research, engineering, and manufacturing. Many of the needed radioisotopes for these applications are present in spent nuclear fuel. The future demand can be met by employing processes in the back end of the fuel cycle to sufficiently separate these radioisotopes. Through such approaches, not only will the quantity of material for disposal be reduced, but also the isotopes in spent fuel that are currently considered to be a waste may in fact become a resource.

Diverse Energy Product Systems

Diverse energy product systems may provide hydrogen, hot water for district heating, seawater desalination and other products in addition to electric power, with or without changing the electrical or thermal efficiency. The idea of direct application of fission energy for hydrogen production or processing of waste heat is not limited to a specific reactor design. With appropriate design the energy product of many power reactor concepts could be diversified.

Modular Deployable Reactor Systems

Modular deployable reactor systems may vary in basic design features but serve the same modular purpose. Modular Deployable Reactors are designed for a long period (5 to 10 years) single cycle use and installation in locations that are not conveniently accessible for construction or refueling. Offshore, underwater deployment is a typical application of such reactors.

Advanced Fuel Materials

Advanced fuel materials include solid solutions of uranium and one or more refractory metal carbides or carbonitrides. Examples of such fuel materials include binary carbides such as (U, Zr)C and (U, Nb)C, tricarbides such as (U, Zr, Nb)C, (U, Zr, Ta)C, (U, Zr, Hf) C, and carbonitrides such as (U, Zr) CN and (U, Ta) CN. These fuel materials feature very high melting point (>3,500 K), very high conductivity (>25 W/m-K), and chemical stability at elevated temperatures. Mixed carbide fuels with niobium or tantalum in place of zirconium, increase the thermal neutron capture cross-sections for these fuels and they are intended exclusively for use in epithermal or fast spectrum reactors. Replacing uranium dioxide or uranium carbide with mixed uranium carbides or carbonitrides in any reactor design would result in significant reduction in average fuel temperature and residual heat content. Reduction of average fuel temperature and heat content improves fission product retention and decreases the probability of a core melt accident.

Alternative Power Conversion Cycles

Due to its technical maturity and commercial availability, the steam turbine (Rankine) cycle has been used in every major nuclear power plant. However, the steam Rankine cycle severely limits the high temperature capability of many reactor concepts. Alternative Power Conversion Cycles include all non-steam turbine cycles with a potential for improved matching to the nuclear heat source. Gas turbine cycle (Brayton) and a few topping cycles, including MHD and thermionics, are proposed to improve the energy conversion efficiency of high temperature reactors.

R&D Scope Report for Nonclassical Reactor Systems

1. INTRODUCTION

The primary goal of the Department of Energy Generation IV Roadmap Development initiative is to identify and evaluate advanced nuclear energy system concepts that offer significant advances towards goals of sustainability, safety and performance, and economy. The primary focus of the TWG-4 is to evaluate novel reactor concepts with design features that put them in a category clearly different from water, gas, or liquid metal cooled reactors. Table 1 summarizes the generic characterizing features of "Classical" versus "Nonclassical" reactor concepts that include Near-Term Deployment, Water Cooled, Gas Cooled, and Metal Cooled Reactors.

Novel nuclear energy system concepts are evaluated based on their potential to improve one or more of the sustainability, safety, and economic goals of Generation IV nuclear energy systems. Overall, the TWG-4 has conducted a global search to identify and evaluate new nuclear energy system concepts. The group has also devoted time to reassess some of the older reactor ideas that were not developed further because of research or commercial priorities. The working group has gathered a total of 32 concepts. After initial screening the TWG-4 identified 28 concepts that meet the Generation IV goals of fission-based self-sustained criticality.

Table 1. Classical versus nonclassical nuclear energy system concepts.

	Classical	Nonclassical
Coolant	Water Gas Liquid/Molten Metals	No coolant Nonconvection coolant Molten salt Organic Two-phase metallic
Fuel Design	Solid/Clad	Solid/No cladding Liquid Gas or Vapor Thin Film
Fuel Cycle	Open Closed, nonintegrated	Fully integrated
Carnot ΔT	1,000–300 K	3,000–300 K ~100 MeV (Direct)
Power Cycle	Steam Rankine Brayton	Direct transformation MHD Thermionic Thermoelectric AMTEC Combined
Applications	Electrical power Medical isotope sources Hydrogen production	Electrical power Medical isotope sources Hydrogen production Chemical refining Space power and propulsion

The evaluation methodology is briefly described following this introduction, but this report focuses primarily on the two sections that describe the nonclassical nuclear energy systems and the crosscut design issues, which together cover all 28 of the viable concepts. The first of these provides the in-depth discussion and evaluation of the six primary concept sets; the second discusses four crosscut design issues that are generically applicable to more than one reactor concept. Four concepts that do not meet the Generation IV definition of an advanced reactor are briefly described in Appendix C.

The references made to sustainability, safety, and economies of the concepts are taken directly from input received by the TWG-4. Although a preliminary evaluation has been conducted, this concept summary report is not intended to reflect the opinions of the technical group with meeting or exceeding Generation IV goals in mind. Furthermore, there are many research and development gaps, feasibility issues, and safety concerns that do not fall within the objectives of this preliminary report and will be included in future concept summary reports.

2. EVALUATION AND CONCEPT SCORING APPROACH

The evaluation process of TWG-4 was conducted in several steps. First, each concept set was assigned to a concept advocate subgroup for in-depth evaluation that included literature search, obtaining additional information from the submitter/author, and discussion among the technical subgroup. The second step included a presentation by the concept advocate subgroup that was followed by an intense question and answer session in which all TWG-4 members participated. Next, a qualitative evaluation in comparison with Light Water Reactors was conducted for each concept with Generation IV performance goals in mind. A qualitative score for every Generation IV goal, ranging from much worse (--) to much better (++), was assigned to each concept by consensus, or at least a majority opinion. A qualitative spread of opinion was drawn out of this evaluation process that was subsequently replaced by a more quantitative measure of a spread of opinion among TWG-4 members. The quantitative measure of the spread of opinion was determined by polling TWG-4 members independently using scores from 1–10 in ascending order with a score of 5 being equivalent to performance of Light Water Reactor systems. Results of this polling were compiled and processed to find the standard deviation for each goal metric. The quantitative approach produced a relatively larger standard deviation than the qualitative approach, mainly due to spreading the scoring range. Due to data uncertainty, the spread of opinion was further exasperated for concepts that are at a lower stage of technological maturity.

3. NONCLASSICAL NUCLEAR ENERGY SYSTEMS

These concepts feature a range of reactor designs with nuclear fuel in gaseous, solid, and liquid phases, with no coolant or with nonconventional coolant, and with a variety of energy conversion systems. A summary description of all 32 concepts is listed in Table 2. Note that the HERACLITUS concept N2 encompasses two concepts, one a molten salt core, the other a molten eutectic metal core. There are four concepts associated with the penultimate category (FSEGT). Two are briefly discussed in Section 3.3.5, the other one (DORC) is a concept for remote deployable reactors and is discussed in Section 4.3.3

Based on the primary design features of the nonclassical nuclear energy system concepts submitted, five “concept sets” are defined as follows:

1. Liquid Core Reactors

2. Gas Core Reactors

3. Nonconventional Coolant Reactors

4. Non-Convection Cooled Reactors

5. Direct Energy Conversion Reactors

Following Table 2 are five subsections (3.1 to 3.5) in which detailed descriptions and tables of evaluation scores are given for each of the five concept sets in turn.

Table 2. Summary of concepts submitted to TWG4 on nonclassical reactor concepts.

Name	No.	Submitted/ Authored	Organization	Country	Type (Category)	Fuel	Coolant	Moderator	Comments on Features
Liquid Core Reactor Concepts									
HERACLITUS (MSR/MMR)	N2a N2b	Palmiotti	Argonne National Laboratory	USA	Molten Salt or Metal (Liquid Core)	Natural thorium fluid	Molten Salt or Metal	Graphite	Circulating fuel, natural thorium.
MSRE	N3	Patton	Marshall Space Flight Center	USA	Molten Salt Core (Liquid Core)	U/Th Salt Fluid	Molten Salt	Graphite	Evolution of Organic Moderated Reactor Experiment.
MSBR	N7	Forsberg	Oak Ridge National Laboratory	USA	Molten Salt Core Pool (Liquid Core)	Liquid U and Th fluorides	Molten Salt	Graphite	Several options for molten salt reactor, including breeders.
LM-FR	N11	McWhirter	Self-employed	USA	Equilibrium Fast Reactor (Liquid Core)	Mg-Pu eutectic	Sodium or NaK	None	Fuel is liquid in a pool with cooling pipes through pool.
MSR	N14	Garzenne	Electricité de France	France	Molten Salt – AMSTER (Liquid Core)	Th, U, or TRU in salt	Molten Salt	Graphite	Graphite moderated molten salt reactor for multiple uses.
Gas Core Reactor Concepts									
GCR/VCR-MHD	N13a	Anghaie	Univ. of Florida	USA	Gas/Vapor Fuel Reactor (Gas Core)	UF_4 vapor	KF or He	To be Determined	Fission enhanced ionization with direct MHD conversion.
GCR-U-C-F	N13b	Kistemaker	Technische Universiteit Eindhoven	The Netherlands	Gas Core-Graphite Wall (Gas Core)	UF_n vapor	CF_4	Water	High T wall corrosion is neutralized.
GCR-UF6	N13c	Lowry	Los Alamos National Laboratory	USA	Vortex Flow (Gas Core)	UF_6 vapor	He	–	Low inventory, zero core meltdown chance.
Plasma Core	N13d	Lowry	Los Alamos National Laboratory	USA	Plasma Vortex Flow (Gas Core)	U vapor, Argon	He	Beryllium	Diverse use, high efficiency.
Nonconventional Cooled Reactor Concepts									
AHTR	N6	Forsberg Pickard	Oak Ridge NL Sandia NL	USA	High Temp. Molten Salt (Nonconventional Cooled)	Graphite matrix	Molten Salt	Graphite	High temp to produce hydrogen and electricity.
OCR	–	Wilson	Univ. of Wisconsin	USA	Organic Cooled (Nonconventional Cooled)	Uranium carbides	Organic compound	Heavy water	Reduced size/costs.

Table 2. (continued).

Name	No.	Submitted/ Authored	Organization	Country	Type (Category)	Fuel	Coolant	Moderator	Comments on Features
FSEGT	–	Trewgoda	Obninsk Inst. of Phys. & Power Eng.	Russian Federation	Liquid Metal Cooled (Nonconventional Cooled)	U or Pu	Sodium Evaporation	Graphite	Unique gas turbine for use with sodium vapor.
FSGT	–	Trewgoda	Obninsk Inst. of Phys. & Power Eng.	Russian Federation	Liquid Metal Cooled (Nonconventional Cooled)	U or Pu	Sodium	Graphite	Unique gas turbine for use with sodium vapor.
FLGT	–	Trewgoda	Obninsk Inst. of Phys. & Power Eng.	Russian Federation	Liquid Metal Cooled (Nonconventional Cooled)	U or Pu	Lead	Graphite	Unique gas turbine for use with lead vapor.
Non-Convection Cooled Reactor Concepts									
Solid State	N15	Klein	Oregon State University	USA	Heatpipe, Solid State (Non-Convection Cooled)	Not specified	Heatpipe, no fluid.	Not specified	Small reactors cooled by conduction & radiation.
Direct Energy Conversion Reactor Concepts									
QSMC	N9	Rochau	Sandia National Laboratory	USA	Fission Fragment Mag. Cell (Direct Conversion)	Fissionable Material	Radiation Cooling	To be Determined	Direct conversion of fission fragments to electricity.
FFMC	N12	Rochau	Sandia National Laboratory	USA	Fission Fragment (Direct Conversion)	Thin film UO_2	Heavy Water	Heavy Water	Direct conversion of fission fragments to electricity.
Waste Minimization Concept (Crosscut Issue)									
CANDLE	N1	Sekimoto	Tokyo Institute of Technology	Japan	Not Specified (Minimize Waste)	Natural uranium	Not Specified	Not Specified	Emphasis on benefits of constant axial burn-up.
MCPP	N17	Alekseev	Russian Research Center (Kurchatov Inst)	Russian Federation	LWR, FR and MSR (Minimum Waste)	Natural U and Th, or Pu	Water, and $^{66}LiF\text{-}^{34}BeF_2$	Graphite	Critical or subcritical MSR to close the fuel cycle.
Diverse Energy Product Concepts (Crosscut Issue)									
Hydrogen	–	Forsberg	Oak Ridge National Laboratory	USA	Any High Temperature (Diverse Products)	NA	NA	NA	Exhausted oil reserves, move to H_2 economy.
Modular Deployable Reactor Concepts									
MMDR	–	Pickard	Sandia National Laboratory	USA	Not specified (Modular Deployable)	NA	Helium	NA	Transportable, factory built, easy assembly.
SPS	–	Herring	Idaho National Laboratory	USA	Not specified (Modular Deployable)	NA	NA	NA	Transportable, modular.

Table 2. (continued).

Name	No.	Submitted/ Authored	Organization	Country	Type (Category)	Fuel	Coolant	Moderator	Comments on Features
DORC	–	Trewgoda	Obninsk Inst. of Phys. & Power Eng.	Russian Federation	Liquid Metal Cooled (Modular Deployable)	U or Pu	NA	Graphite	Remote operated.
Advanced Fuel Materials (Crosscut Issue)									
AFS1	–	Anghaie	Univ. of Florida	USA	Tricarbide & Bicarbide fuels (Advanced Fuels)	(U, Zr, Nb)C	Helium	Graphite	Ultrahigh temp., very high m.p.
AFS2	–	Anghaie	Univ. of Florida	USA	Carbonitride fuels (Advanced Fuels)	(U, Zr)CN	Helium	Graphite	Ultrahigh temp., very high m.p.
Alternative Power Cycles (Crosscut Issue)									
Alternative Cycles	N16	Klein	Oregon State University	USA	Topping cycles (Alternative Power Cycles)	NA	NA	NA	Efficiency gains, reduced plant cost.
REMHD	–	Kadak	MIT	USA	Radiation Enhanced MHD (Alternative Power Cycles)	NA	Helium, cesium	NA	High Carnot efficiency, direct.
Others (Non-Generation IV Concepts)									
TASSE	N4	Palmiotti	Argonne National Laboratory	USA	Accelerator Driven (Other)	Thorium	Molten Salt or Metal	Not Specified	Subcritical, thorium fuel, feed and bleed.
ENR	N5	Matsumoto	Hokkaido University	Japan	Electro-Nuclear Reactor (Other)	Water	Not Specified	Not Specified	Revolutionary new electro-nuclear reactions.
CBNS	N8	Miley	Univ. of Illinois	USA	Neutron Source-Subcritical (Other)	Not Specified	Liquid Metal	None - Fast Reactor	Neutron source driven, subcritical.
ADSH2	N10	Park	Korean Atomic Energy Res. Inst.	Korea	Accelerator Driven System (Other)	TRU-Zr Fuel	Lead-Bismuth	Not Specified	Generate H_2 by transmuting TRU from LWR spent fuel.

Table Legend:

N2a “High Efficiency, Reactivity At constant Level, Intrinsically-safe, Thorium-fueled, Unpolluting System” (HERACLITUS) —Molten Salt Core.

N2b HERACLITUS—Molten Metal Core.

N3 “Molten Salt Reactor Experiment” (MSRE).

N7 “Molten Salt Breeder Core Reactor” (MSBR).

N11 “Liquid Metal Fast Reactor” (LMFR).

N14 “Molten Salt Core Reactor.”

Table 2. (continued).

Name	No.	Submitted/ Authored	Organization	Country	Type (Category)	Fuel	Coolant	Moderator	Comments on Features
N13a	"Gas or Vapor Core Reactor with Magnetohydrodynamic Converter" (GCR/VCR-MHD).								
N13b	"Gas Core Reactor Graphite Wall U-C-F system with Magneto-inductive Piston Converter" (GCR-U-C-F).								
N13c	"Mixed Vortex Flow UF6 Gas Core Reactor" (GCR-UF6).								
N13d	"Plasma Vortex Flow Gas Core Reactor" (Plasma Core).								
N6	"Advanced High Temperature Reactor" (AHTR)—modular molten salt cooled, graphite matrix fueled AHTR.								
–	"Organic Coolant Reactors" (OCR)—evolution from the CANDU-OCR reactor.								
–	"Fast Sodium Evaporation and Gas Turbine" (FSEGT)) —molten metal–gas turbine (GT) concepts— (NB: should belong to TWG3).								
–	"Fast Sodium–Gas Turbine" (FSGT— (NB: should belong to TWG3).								
–	"Fast Lead-cooled–Gas Turbine" (FLGT)— (NB: should belong to TWG3).								
N15	Solid State, Heatpipe-Cooled Reactors.								
N9	"Quasi-Spherical Magnetic Cell" (QSMC)—fission electric cell array.								
N12	"Fission Fragment Magnetic Collimator" (FFMC)—fission electric cell.								
N1	"Constant Axial shape of Neutron flux, Nuclide densities and power shape During Life of Energy production" (CANDLE).								
N17	"Multi-Component Power Plants" (MCPP)—molten salt core reactors as part of multi-component nuclear power plant.								
–	"Hydrogen"—submission on hydrogen production as a major application of Nonclassical nuclear reactors.								
–	"Multi-Modular Power Reactors" (MMDR) —no specific reactor.								
–	"Submersible Nuclear Power Station" (SPS) —no specific reactor.								
–	"Distantly Operated Reactor Concept" (DORC) —conceived along with the other molten metal–gas turbine (GT) concepts.								
–	"Advanced Fuel Systems" (AFS1)— high melting point bicarbides and tricarbide fuels for high performance/high temperature reactors.								
–	"Advanced Fuel Systems" (AFS2)— high melting point carbonitrides for ultrahigh temperature gas-cooled reactors.								
–	"Radiation Enhanced MagnetoHydroDynamic power" (REMHD).								
N16	Alternative Power Cycles—no specific reactor.								
N4	"Thorium-based Accelerator-driven System with Simplified fuel cycle for long-term Energy production" (TASSE).								
N5	"Electro-Nuclear Reactor" (ENR).								
N8	"Converging Beam Neutron Source" (CBNS)—subcritical reactor.								
N10	"Accelerator Driven System" (ADSH2)—for hydrogen production.								

3.1 Liquid Core Reactors

Number	Concept Name	Sponsorship
N2	HERACLITUS—Molten Salt and Molten Metal	ANL
N3	MSRE—Oak Ridge Experiment evolution	NASA Marshall SFC
N7	MSBR—Molten Salt Breeder Reactor	ORNL
N11	LMFR—Liquid Metal Fast Reactor	McWhirter
N14	AMSTER—Actinides Transmuter	EdF

The Liquid Core refers to any reactor in which the fuel in the core is in the liquid phase. The Liquid Core Reactor concept set includes two subsets: molten salt fuel and eutectic molten metal fuel reactors. Before describing these, it is worthwhile listing the main benefits of changing from solid to liquid fuels. These are: (1) the ability to enhance thermal efficiency through high temperatures, (2) the high temperatures achievable with liquid core reactors have the potential of creating new applications for nuclear power beyond electricity generation, (3) simplification of the nuclear power plant design (no fuel assemblies, control rods, assembly support and discharge systems, etc.), (4) elimination of reactivity shuffling during operation, (5) on-line fuel treatment (if needed) capability, (6) ability to work with a high concentration of fission products (corresponding to the high burn-up requirement), (7) intrinsic safety, and (8) drastic reduction in the probability of reactivity-induced accidents.

3.1.1 Molten Salt Core Reactor Concepts

Molten Salt Core Reactors (MSRs) were initially developed in the early 1950s as part of the Aircraft Nuclear Propulsion Program in the United States. A small test reactor was successfully operated. While the concept of a nuclear powered aircraft proved unfeasible because of shielding weight, the research indicated the potential to build a high thermal efficiency, thermal neutron ^{232}Th-^{233}U breeder reactor for power production. The molten salt reactor experiment (MSRE), an 8 MW_{th} test reactor, demonstrated the concept for energy production and was successfully operated from 1965 to 1969 at Oak Ridge National Laboratory.[1] The development program[2,3] and results from operation of the test reactor resulted in a detailed conceptual design of a 1,000 MW(e) Molten Salt Breeder Reactor (MSBR) that became the backup to the fast-breeder reactor program in the United States. Table 3 and Figure 1 show the characteristics of this reactor.

With a national decision to concentrate resources on a single breeder reactor, the MSR development program was shut down in the early 1970s, except for a small research program to address key technical feasibility issues. When this decision was made, uranium resources were thought to be extremely limited. The fast reactor was preferred because of its higher breeding ratio and lower doubling time. The molten salt breeder reactor has a maximum breeding ratio of ~1.06. This is significantly below that possible in a fast reactor.

As a consequence of changing requirements and advances in technology, a number of countries and different organizations are re-examining MSRs. The proposed new designs reflect new technologies that have become available and modifications to meet different goals. The French utility, Electricité de France (EdF), is examining MSRs for power production and partitioning/transmutation of higher actinides.[4] This effort includes AMSTER (Actinides Molten Salt Transmuter). The Kurchatov Institute[5] in Russia has proposed a nuclear energy future with three reactors: light-water reactors for power, fast reactors for fissile material production and power, and MSRs for burning higher actinides and power. Argonne National Laboratory, with cooperation of the French, has proposed HERATICLUS[6] (High Efficiency,

Table 3. Characteristics of a large molten salt reactor.

Net Electric Power	1,000 MW	Maximum core flow velocity	2.6 ms^{-1}
Thermal efficiency	44.4%	Total fuel salt	48.7 m^3
Core height	3.96 m	^{235}U	1,500 kg
Vessel design pressure	5.2×10^5 Nm^{-2} (75 psi)	Thorium	68 100 kg
Average power density	22.2 kW per L	Salt components	$^7LiF–BeF_2–ThF_4–UF_4$
Graphite mass	304,000 kg	Salt composition (per above)	71.7–16–12–0.3 mol%

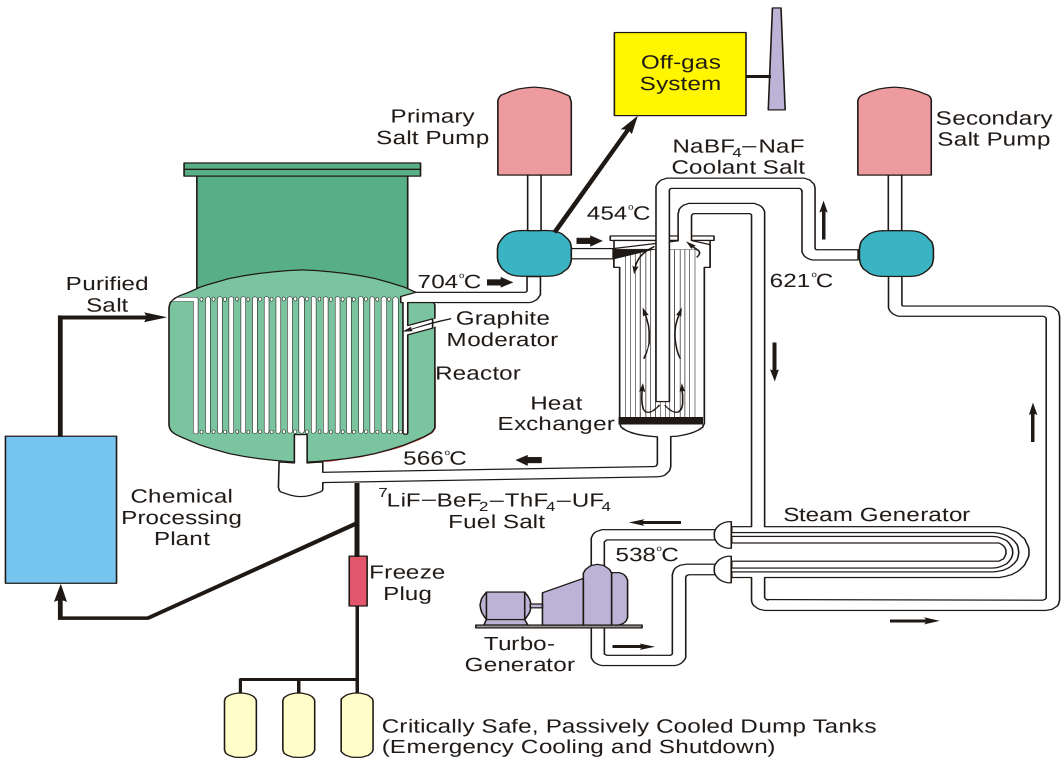


Figure 1. Layout of a Molten Salt Core Reactor power plant. (Drawing courtesy of ORNL.)

Reactivity at Constant Level, Intrinsically-safe, Thorium fuelled, Unpolluting System). HERATICLUS has been proposed with two alternative fluid fuels: molten salt and molten metal. Japanese researchers have also made several proposals.[7] ORNL has proposed changes to improve proliferation resistance.

The basic characteristics of the MSR remain unchanged. The fuel is a liquid mixture of lithium-7 fluoride, beryllium fluoride, thorium fluoride, and uranium fluorides. Other salt mixtures are possible. However this particular salt was proven during the MSRE and offers the best neutron economy and breeding performance. During operation, most fission products and all actinides form fluorides in the liquid. The liquid fuel salt flows upward through vertical channels into the unclad graphite core. The graphite moderates the epithermal spectrum that is characteristic of the salt, such that criticality occurs only in the reactor core. The concentration of fissile materials in the salt is limited by their solubility. As a consequence, the neutron moderator characteristics of the salt are sufficient that a fast-spectrum molten salt reactor is not possible. The heat is generated directly in the molten fuel.

MSRs can have either an intermediate or thermal neutron spectrum. A fast-spectrum fluoride molten salt reactor is not possible. Fissile concentrations in the salt are limited by solubility limits. The ratio of fluoride to fissile atoms is sufficiently high that the fluoride atoms provide significant neutron moderation.

The fluid fuel flows through a primary heat exchanger, where the heat is transferred to a secondary coolant, and back to the reactor core. The heat is dumped from the secondary system to the power cycle. The original MSBR proposed using a molten salt secondary loop and a high-temperature steam cycle. More recent proposals, such as those by EdF,[8] have suggested using a helium gas turbine power cycle. Molten salt melting points are above 400°C, thus all MSRs operate at relatively high temperatures with high thermal efficiencies.

There are two fluid-fuel cleanup systems. A high-efficiency gas-stripping system incorporated into the primary circulation pumps removes noble gases (xenon, krypton, etc.) and tritium. The noble gases, particularly certain xenon isotopes, are strong neutron absorbers. Without the quick removal of the gases, the neutrons absorbed by these gases would prevent the reactor from being a breeder reactor. A salt-cleanup system removes lanthanides and other fission products from the salt and controls the salt composition. Much of the recent work on MSRs by EdF and others has been to simplify this cleanup system. The goal is no longer to maximize the breeding ratio but rather achieve a breeding ratio near 1. This allows for simpler salt processing systems and provides other advantages.

There are four major fuel cycle options, which can be used with particular intended goals in mind. (1) maximum breeding (breeding ratio ~1.06), (2) denatured thorium-^{233}U breeder with minimum inventory of weapons-usable material,[3,9] (3) denatured "once-through" fuel cycle with minimum chemical processing and (4) actinide burning.[10] The reactor can use ^{238}U or ^{232}Th as a fertile fuel, however, ^{232}Th must be the primary fertile material if the reactor is operated as a breeder reactor. All of the MSRs can be started up using low-enriched uranium or other fissile materials.

3.1.1.1 Generation IV Goals—Capabilities for Molten Salt Core Concepts.

3.1.1.1.1 Sustainability Aspects—*Sustainability 1*—The MSR can operate as a thermal breeder reactor on a ^{232}Th – ^{233}U fuel cycle, with very low resource demands. The production of 1 TWhr requires about 20 tons of natural uranium in case of a pressurized water reactor (PWR), (with a burn-up of 60 GWday/ton) compared to 100 kg of natural thorium for a MSBR. Alternatively, the MSR can be refueled with a mixture of ^{232}Th and some ^{238}U with a lower breeding ratio. The MSR has a low combined reactor and fuel cycle fissile inventory for three reasons: (1) it is a thermal neutron reactor with a small critical mass, (2) it has a high power density in the liquid, and (3) there is no external fuel cycle fissile inventory after reactor startup. The fuel cycle impact on the environment is low because the fissile and fertile requirements are low. The utilization of other resources is expected to be similar to a light water reactor (LWR).

Sustainability 2—The MSR minimizes waste output. The thorium fuel cycle minimizes the generation of higher actinides. In the normal mode of operation, fission products are removed from the molten salt but actinides remain until they are fissioned. The current estimates for the French AMSTER concept indicate a deposit of about 20 grams of transuranic elements in the wastes to the repository for each TWhr produced, compared to 30 kg per TWhr for an LWR with an average burn-up of 60 GWday/ton.

Because of their unique characteristics, MSRs are being evaluated in several countries for transmutation of actinides from other reactors. A molten salt core reactor does not require fuel fabrication, which is a very expensive and difficult process for the higher actinides (americium and curium). There is no recycling of actinides out of a MSR once they are added to the molten salt. The dilute concentration of actinides in the salt avoids the handling of concentrated higher actinides with their very high decay-heat generation rates. The inventory of actinides required to maintain a critical MSR is low compared to other reactors. The alpha activity of the higher actinides is several orders of magnitude greater than the activity of plutonium. This implies a large accident source term in reactors with high fissile fuel inventories per unit of power generation.

Sustainability 3—The nonproliferation characteristics of a MSR are defined as equivalent to an LWR. When the early work on MSRs was conducted, nonproliferation was not an important consideration, thus it was not studied. The characteristics of the MSR are radically different than other reactor concepts, and there has been only very limited work to understand the nonproliferation characteristics of this reactor. There is the potential that MSR proliferation resistance could be significantly better than LWRs. The inventories of weapons-usable fissile materials can be made very low compared to other reactors. The fissile material isotopics are unusual and unfavorable for use in weapons. ^{242}Pu is the primary plutonium isotope in the MSR whereas ^{239}Pu is preferred for weapons. The ^{233}U can be made nonweapons usable by keeping a sufficient inventory of ^{238}U in the reactor. All fissile materials are contained inside the reactor and associated processing units. There is no significant fissile flow outside the reactor (transportation, repository, other). The newer MSR concepts, such as being developed in France, have low processing rates for the salt that imply long periods of time to remove significant quantities of fissile materials from the reactor. These issues are further discussed Appendix A.

3.1.1.1.2 Safety and Reliability Aspects—*Safety and Reliability 1*—There are several intrinsic characteristics that minimize the potential for an accident. The molten fluoride salts are chemically stable at high temperatures. The temperature margin to the salt boiling (1,400°C) is large. The thermal inertia (total heat capacity) is large because of the large amount of graphite moderator inside the core. The online feeding and processing of fuel minimizes the need for excess reactivity. Reactivity control is achieved by a prompt negative temperature coefficient due to the Doppler effect and the thermal expansion of the fluid fuel. EdF estimated the MSBR temperature coefficient to be -0.9 pcm/°C, resulting in a much stronger coefficient of -3.3 pcm/°C for the fuel itself, and a positive graphite coefficient of +2.4 pcm/°C. Thanks to its thermal inertia, the graphite changes temperature much more slowly than the fuel salt, and consequently heating of the fuel results in a prompt negative response of reactivity to a power excursion.

Safety and Reliability 2—MSRs use passive cooling systems that can be scaled to any size of reactor. In the event of an off-normal condition and during maintenance shutdowns, the molten salt is drained to a passively cooled, critically safe storage tank, Figure 2. Several proposed designs include overflow pipes from the core to storage tanks. If the fuel overheats and expands, the excess fuel is dumped to these tanks. The molten salt test reactors used freeze valves that froze salt in a section of the drainpipe. Loss of refrigeration dumped the salt to the storage tanks. The more recent EDF design proposes to pressurize the drain tanks using a helium compressor to force the salt into the primary system. If power to the compressor is lost, the storage tanks would depressurize and the molten salt would dump to the drain tanks.

The primary system is located in an inert-gas, sealed hot cell that is a secondary containment system. If there is a leak from the primary system, the molten salt drains from the hot cell into the passively cooled drain tanks. Because of economic, safety, reliability, and maintenance benefits, the hot cell (not the reactor vessel or individual pipes) is thermally insulated. This avoids the potential of a localized insulation failure that could cause local salt freezing. The test reactors and many of the proposed power plants would use a confinement system or containment as a third barrier to radionuclide release.

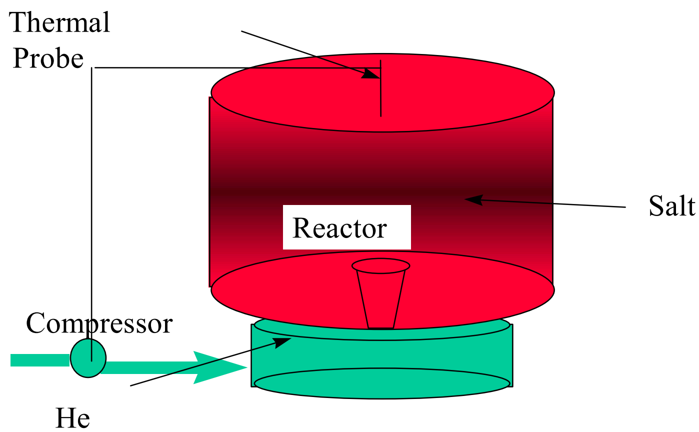


Figure 2. Core drainage principle for molten core reactors. (Drawing courtesy of ANL.)

Safety and Reliability 3—There are several mechanisms that minimize the potential release of radionuclides under accident conditions: (1) the MSR operates at a low-pressure with minimal driving force for radionuclide release, (2) the accident source term is lower than any other power reactor (online "nuclear dialysis" removal of selected fission products minimizes volatile fission products that could escape in an accident), and (3) the fission products and actinides are soluble in a molten salt that freezes at about 400°C. Recent Russian studies indicate that the total release of radioactivity in a serious accident would be one to two orders less than a LWR because of the above factors. However, to minimize total risks (not just transfer part of the source term to a secondary process unit), the radionuclides removed from the molten salt are solidified and placed into passively cooled storage systems.

3.1.1.1.3 Economic Aspects—Several characteristics of MSRs suggest the potential for good economics. The high temperature (>700°C) of the nuclear heat source allows for high efficiency power conversion (up to 44%) as well as the potential for producing other energy products such as hydrogen. The primary system is a low-pressure system that allows reactors of very large size to be built and reduces reactor, containment, and other costs. The MSR has a simplified fuel cycle. There is no fuel element fabrication or recycling required and no transport of highly radioactive materials except for high level waste after storage to reduce decay heat loads. Very large reactors can be built with passive safety systems. There are potentially large economic gains for sites with multiple reactors where common off-gas, salt processing, and maintenance facilities can be used.

Despite the apparent economic gains afforded by MSRs relative to LWRs, there are several characteristics that add costs. A MSR requires remote maintenance of the primary system and many of the auxiliary systems. Many systems are almost independent of plant size. This indicates significant economics of scale unless several small reactors are co-located and share common services.

The one economic analysis, based on comparable designs for a PWR and a MSR in ~1970, indicated that MSR economics were superior to an LWR for a 1,000 MW(e) machine. However, goals have changed and there are no current detailed designs available to allow an updated cost estimate.

3.1.1.1.4 On Institutional Dependencies—MSRs have unique institutional characteristics. Almost all fuel cycle operations are at the reactor site. The wastes would likely be stored at the site for decades to allow reduction in decay heat before disposal in the repository. This minimizes transportation, transportation risks, the need for security anywhere except at the reactor site, and other offsite constraints. However, it does require the power plant owner to have capabilities to maintain and operate hot facilities.

3.1.1.2 Research and Development Challenges—MSR. The following key issues associated with MSRs are divided into two categories: (1) research areas where the technology appears to exist but where added research, development or testing is required to have high confidence that issues are resolved and (2) research areas where there are strong incentives to find new approaches. In the first category are:

Corrosion Resistance—The reactor experiments were successful, but potential long-term corrosion problems caused by the fission product tellurium were identified. Modified alloys were then developed that in laboratory tests met almost all requirements. Added testing, including reactor test loops, is required to have full confidence in the alloys.

Tritium Control—The ^{7}LiF in the salt results in significantly higher tritium production than in other reactors. At the high temperatures found in a MSR, the tritium can diffuse through the heat exchangers to the secondary system. The ORNL designs of MSRs used sodium fluoroborate as a secondary cooling salt to chemically trap the tritium. French CEA studies indicated tritium diffusion from the primary system could be stopped by counter diffusion of small quantities of hydrogen from the secondary loop salt. The tritium control technologies need to be fully developed and tested under a wide range of conditions.

Off-gas System—To meet future safety goals, the off-gas system must quickly immobilize captured fission products so the accident risks are not just transferred from the reactor to the off gas system but are significantly reduced. Some but not all of these technologies have been developed for the MSR.

Current Regulatory Design—The current regulatory structure is designed for solid fuel reactors. Significant changes in design may be required to meet the intent of current regulations. Work is required with regulators to define equivalence in safety for a reactor with very different characteristics.

Design and Economic Evaluation—No detailed design of a MSR has been done since 1970. An updated design (including design tradeoff studies) is required to better understand strengths and weaknesses and allow credible economic evaluations.

Below are two areas where significant research is required:

Non-proliferation—When the original MSR technology was developed, nonproliferation was not a concern. MSR fuel cycles are fundamentally different than once-through LWR fuel cycles and traditional closed fast reactor cycles (see Appendix A). Research is required to understand the issues and determine if design changes are required.

Salt Processing—The early work on salt processing developed and demonstrated flow sheets on a laboratory scale to remove radionuclides from the salt to maximize the breeding ratio. No work was done to convert the wastes into acceptable forms for disposal. At the time no disposal criteria had been defined. Goals have since changed, which now require that: (1) wastes should be rapidly immobilized into high-quality waste forms to meet repository requirements and minimize on-site safety risks under accident conditions, (2) a breeding ratio near one is acceptable that, in turn, allows flow sheet simplification, and (3) MSRs are being considered for transmutation of actinides from other reactors. The changes in goals combined with advances in salt processing indicate research is needed in salt processing.

Molten salt technology is being developed for other applications. The same molten salts are the leading candidates for cooling fusion power reactors. There is an interest in space molten salt reactors for production of electricity. MSRs were originally developed for the Aircraft Nuclear Propulsion Program because modified designs can have very high power to weight ratios compared to other reactors. Many of the selection criteria for space reactors are similar to those for an aircraft reactor. These related applications imply a potential for synergic R&D.

3.1.2 Eutectic Metal Core Reactors

Eutectic metallic fuel reactors use a mixture of uranium or plutonium and a low neutron absorbing metal. A separate heat transport system is needed to remove energy from the core, for which helium is the working fluid of choice. Iron is a good solvent metal if moderate temperatures are desired. However iron also has a high neutron capture cross section that degrades the neutron economy, therefore aluminum has also been considered as an alternate fuel solvent. As with molten salt cores, molten eutectic metal reactors can achieve control of criticality by nuclear dialysis that removes fission products online. The fertile fuel is also added via the external online circulation system. The eutectic metallic fuel concept is in very early stage of development. Two specific concepts are described below.

3.1.2.1 HERACLITUS—Molten Metal Core Reactor (MMR). This is a molten metal version of HERACLITUS under study at ANL.[6] The reactor has a converter core and a fast spectrum. A preliminary model for a fast spectrum aluminum liquid metal fuel reactor is shown in Figure 3. Surrounding the cylindrical container made of refractory metal (tantalum or W-Re alloy) is a solid stainless steel reflector. Among the different possible solvents considered, iron is often chosen because of the eutectic between iron and the heavy metals that makes it possible to achieve the required solubility at reasonable temperatures. However, iron has a significant capture cross-section that is detrimental to the neutron economy. Aluminum is often considered because aluminum is more benign to the neutron economy than iron and allows the elimination of structural material inside the core, and a significant reduction in the total heavy isotope inventory is made possible.

The illustrated configuration has been drawn for an aluminum solvent version. The refractory metal is niobium and the power conversion, as in the case of the MSR, is located outside the core. For the same characteristics as the iron option, the critical volumetric fractions, keeping a conversion ratio close to one, are: U^{233} 1.6%, Th^{232} 22%, and Al 77.4%. For an iron metal solvent version the volumetric fractions

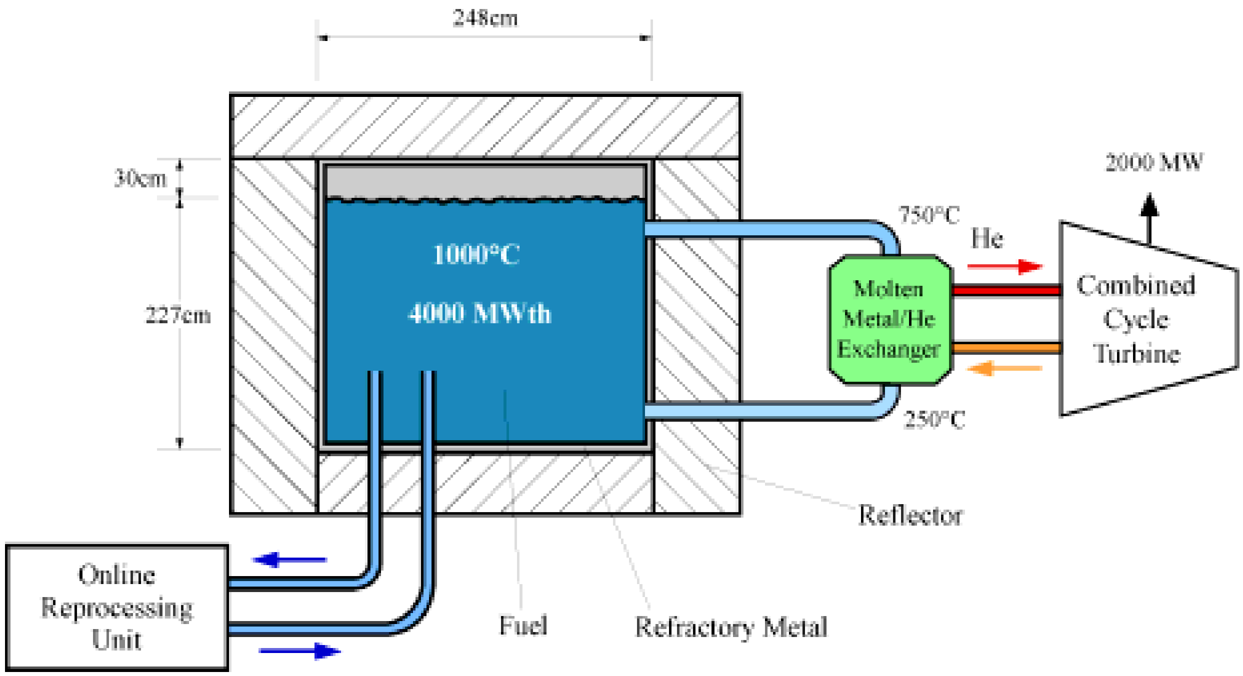


Figure 3. HERACLITUS molten metal core reactor based on aluminum solvent. (Drawing courtesy of ANL.)

inside the container are 2.8% of U^{233}, 50.2% Th^{232}, 14% of Fe and 33% of coolant plus piping (the iron molt would need Helium cooling U-tubes running through the core).

3.1.2.2 Generation IV Goals—Capabilities for Molten Metal Core Concepts.

3.1.2.2.1 Sustainability Aspects—As an option the long-lived fission products can be transmuted in situ to further reduce the radiotoxicity risk.

3.1.2.2.2 Safety and Reliability Aspects—HERACLITUS inherits many of the safety aspects of molten core reactors. The core can be drained in an emergency by a passive helium backpressure plug (Figure 2). A self-heating coolant combined with self-limiting reactivity feedbacks and enhanced natural circulation capabilities promote fission power control and heat convection to the heat sink in even the most severe accident sequences. Furthermore, criticality is maintained by nuclear dialysis that removes the fission products on-line.

3.1.2.2.3 Economy Aspects—HERACLITUS benefits in its conception from many economical advantages. The simplification both in terms of components (no fuel assemblies, control rods, assembly support and discharge systems) and fuel cycle combined with the increase in efficiency promises to deliver a system that will generate electricity at a rate competitive with the cheapest fossil fuel sources (e.g., natural gas) currently used in power generation.

3.1.2.3 Liquid Metal Fast Reactors. The family of reactor concepts[11,12] distinguished by the use of a fluid core in the form of a molten metal all operate as fast reactors like the Heraclitus MMR version. They can either consist of pools with in-core heat exchangers, or some of the fluid might circulate to an ex-core heat exchanger. This is expected to allow for higher thermal efficiencies (due to higher operating temperatures), higher burn-ups/fuel utilization and fuel cycle flexibility for the minimization of divertible fissile material. A Liquid Metal fast reactor that uses temperature gradient induced circulation is depicted in Figure 4. The coolant should ideally be nonreactive with the fuel, as well as immiscible, should a leak between the coolant system and the fuel occur. Coolants such as sodium and NaK (eutectic sodium-potassium alloy) match both of these criteria.

The nuclear performance of the system should closely mimic that of a sodium-cooled fast reactor in terms of neutron energy spectrum. However, magnesium has an even smaller absorption coefficient than sodium or potassium. The reactor system would be designed to be self-regulating and inherently stable. During power transients or a loss of coolant flow, the overall system temperature increases, the fuel expands, and negative reactivity is added because of increased boiling (if the geometry is properly designed). Inherent stability occurs since boiling within the core causes a concomitant reduction in the fuel concentration at the point of bubble formation.

3.1.2.4 Generation IV Goals—Capabilities for Liquid Metal Fast Reactor Concepts. On the issues of safety, sustainability and economy, the Liquid Metal Fast Reactor fairs quite favorably.

3.1.2.4.1 Sustainability Aspects—In situ breeding of plutonium greatly reduces the need for feeding enriched fuel, and retention and irradiation of the soluble and insoluble fission products is believed to substantially reduce the total radioactive waste efflux. Weapons proliferation would be extremely difficult because the reactor functions as plutonium burner and so would be extremely dilute and contaminated with fission products.

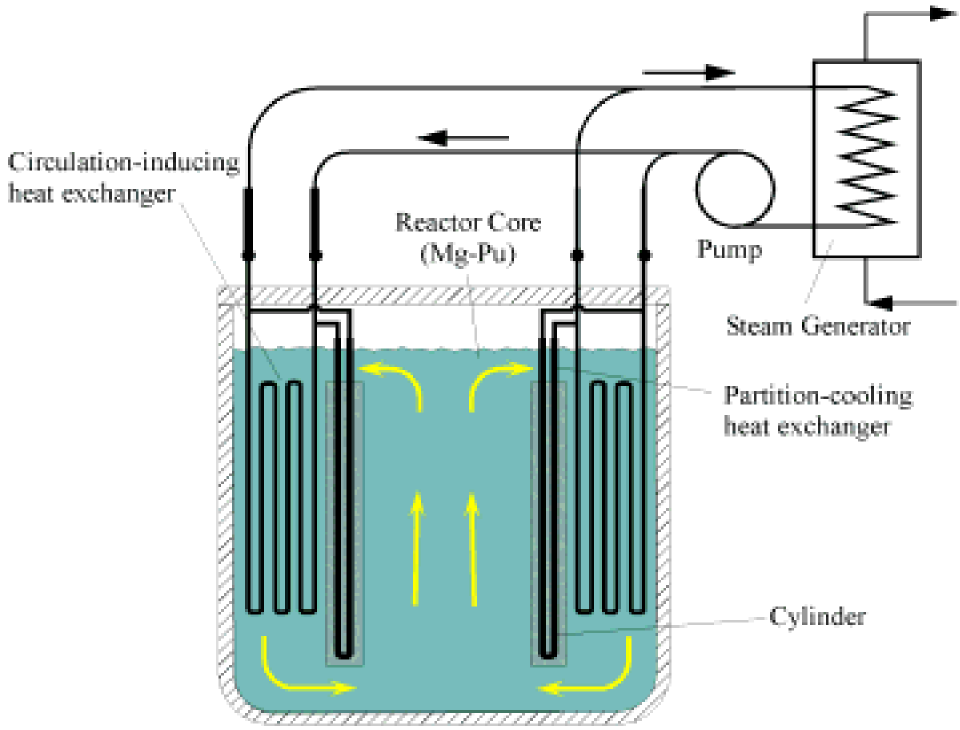


Figure 4. Liquid Metal Fast reactor schematic. (Drawing courtesy of Jon McWhirter.)

3.1.2.4.2 Safety and Reliability Aspects—The proposed concept is very simple and has a natural circulation of fuel, so the fewer mechanical parts add to safety and reliability. With the liquid core there is no core damage criteria to worry about except leakage, but cleanup would be expeditious owing to the designed immiscibility of the coolant with the fuel. Offsite emergency responses would be avoidable because most of the fission product inventory is in gaseous and volatile form and can be removed in situ.

3.1.2.4.3 Economic Aspects—The simple design helps reduce capital costs as well as fuel costs. Decommissioning and decontamination costs are expected to be low because the fuel mixture would be transferred to another reactor facility for use, leaving only a few parts left to quickly and cheaply decontaminate. The financial risk is expected to be manageable again due to the overall simplicity of the scheme. Modules could be set up to supply-heated coolant to a single steam generator or other conversion process. The gaseous/volatile fission product cleanup systems could also be shared between several plants.

3.1.2.5 Research and Development Challenges—MMR. The primary challenge of designing the HERACLITUS system will be due to the use of new metal and refractory materials at high temperatures with a clear constraint to reduce capital investment, achieve long operating life, and reduce downtime due to the need for replacing components. Challenges common to all liquid core concepts include the increased possibility of leaks, pumping power and pump chemistry concerns, and increased opportunity for diversion of material. The benefits of online chemical processing needs to be weighed against the potential proliferation risks. In some cases investigation of possible improvements offered by new technologies and generally more economic and reliability analyses need to be performed.

3.1.3 Multicomponent MSR Systems

Several countries are investigating nuclear futures that combine several types of reactors. In many of these systems, MSRs are chosen as the preferred option for the destruction of actinides to minimize wastes to the repository. An example is the Russian Kurchatov Institute proposal (see Figure 14) for a system containing (1) fast reactors to produce fuel and power, (2) light water reactors for power, and (3) MSRs for power and destruction (transmutation) of higher actinides. MSRs are added to the system for two purposes: (1), their unique characteristics may provide the most economic method for actinide destruction and (2), the complexity of burning higher actinides in solid-fuel fast or light-water reactors may degrade the economics or safety of these reactors. Various designs for such multcore power systems are discussed later in Section 4.1.1 under the topic of "Minimum Waste Fuel Cycles."

3.1.4 Concept Viability Evaluation for the Molten Salt Core Reactors

The score sheet developed through a detailed concept presentation and discussion within the Nonclassical Concept TWG is presented in Table 4. A previous TWG-4 meeting established the score sheet, based upon consensus. It is intended as a qualitative screening of the concept's potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

The decision to retain the MSR concept for further Generation IV evaluation was arrived at after considerable deliberation by all members of TWG-4, and fully takes into account all the strengths and weaknesses of the liquid core reactor concepts.

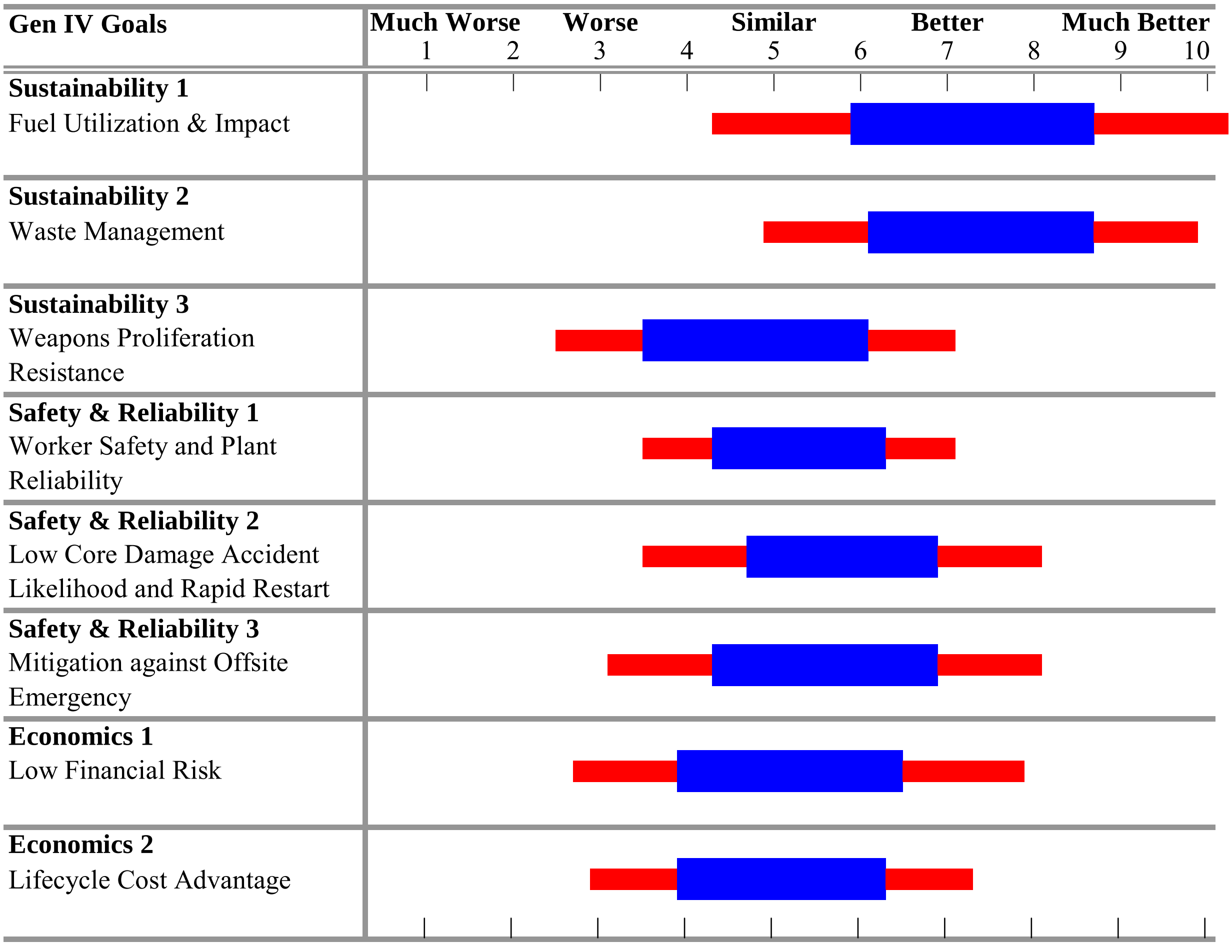
In addition, results of independent polling/evaluation of the members the TWG-4 are presented in Table 5. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean. Note that the mean scores in this table are close to the respective qualitative scores of Table 4 for the respective lumped goal categories. Under each goal there were a number of criteria and metrics, all of which were scored, but only the lumped averages are presented here.

The large spread of scores indicated in this table reflects both the uncertainty in the data that was available for the 14 TWG-4 members to base their assessments upon, as well as the expanded scale that now ranges from 1–10, rather than the five-fold (– –,–,=,+,++) scale used in the qualitative consensus-based screening.

Table 4. Qualitative group summary evaluation—liquid core set.

Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization and Impact	++	• Thorium breeder reactor • Low fissile inventory in reactor and fuel cycle • All fissile materials fully utilized
Sustainability 2 Waste Management	++	• Low actinide inventory (^{233}U/thorium cycle) • Actinide burner with low actinide waste
Sustainability 3 Weapons Proliferation Resistance	=	• Low weapons usable fissile inventory • Natural uranium & thorium feed (after startup) • High burn-up isotopics (^{242}Pu, etc.) • Limited analysis on unusual fuel cycle
Safety and Reliability 1 Worker Safety and Plant Reliability	+	• Natural circulation • In situ waste removal (nuclear dialysis) • Low pressure
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	+	• No fuel damage, only leakage worry • Core plug drainage (dump fuel to passively cooled drain tanks)
Safety and Reliability 3 Mitigation against Offsite Emergency	+	• Low accident source term (in situ removal of fission products) • Low pressure
Economics 1 Life-Cycle Cost Advantage	+	• High electric conversion efficiency • Economics of scale or co-siting of plants • Low pressure (systems, containments, etc.) • Remote operation and maintenance
Economics 2 Low Financial Risk	+	• Significant technology base but limited experience, technical uncertainties • Fuel cycle within plant—not sensitive to changes in external fuel cycle costs • Actinide destruction with fewer waste management financial risks

Table 5. Quantitative score sheet evaluation for the Liquid Core Reactor Group.



3.2 Gas Core Reactors

Number	Concept Name	Sponsorship
N13a	GCR/VCR-MHD — Gas Core and Vapor Core	University of Florida
N13b	GCR-U-F-C—Gas Core Graphite Wall	Technische University. Eindhoven
N13c	GCRUF6—Gas Core Vortex Flow	LANL
N13d	Plasma Vortex Flow Gas Core	LANL

The Gas Core Reactor concept set[13,14] comprises all reactors that are fueled by stable uranium compounds that are in gaseous phase at sustainable engineering temperatures. The primary fuel for these reactors is uranium tetrafluoride, which is one of the most stable uranium compounds in liquid and gaseous phases. A gas core reactor with a condensable fuel such as uranium tetrafluoride is commonly referred to as a Vapor Core Reactor (VCR). A condensable or non-condensable working fluid is added to enhance the heat transport property of the flowing core material. These gas core reactors feature power generation and conversion at very high temperatures. The heat is used in stages. A closed magnetohydrodynamic (MHD) power generation cycle is first used to directly process and convert fission power at temperatures of 1,800–2,500 K. With the rejected heat from the MHD power cycle used to power single or multiple gas turbine and/or superheated steam cycles. The overall efficiency of the Gas

Core Reactor systems is in excess of 60%. But in addition to electricity production, the high temperatures that gas core reactors intrinsically operate at will find many applications that are not options for conventional nuclear reactors, such as hydrogen production and new chemical process applications requiring high temperature. Four specific design concepts are described below. One specific concept, the GCR/VCR-MHD system is evaluated according to Generation IV criteria. Safety, sustainability, and economic advantages are outlined, and finally considerations of technological maturity are given.[15]

3.2.1 The Coupled GCR/VCR-MHD Concept

The VCR-MHD/Brayton/Rankine concept is now in the process of further evaluation. One particular scheme for GCR-MHD power conversion is shown in Figure 5. The single most relevant and unique feature of gas/vapor core class reactors is that the reactor outlet temperature is not constrained by solid fuel-cladding temperature limitations, and is only constrained by the vessel limits, which is far less restrictive. It combines the functions of fuel and coolant into one. Therefore, VCRs can potentially provide the highest reactor and cycle temperature among all existing or proposed fission reactor designs.

Basic design characteristics of a 1,000 MWe UF_4/KF vapor core reactor with direct MHD and indirect Brayton and superheated Rankine cycle are listed in Table 6.

Three key features can be identified, volumetric energy conversion, fission enhanced conductivity, and high Carnot efficiency. Volumetric energy conversion is an extremely promising process because it allows the direct use of the energy generated in the core, at its highest quality, before it is used in a heat cycle. The limiting constraint for efficient use of nuclear power is the maximum effective temperature available for energy conversion. For solid fueled reactors, the fuel melting and the associated cycle materials temperatures dictate these limits. Presently, the most promising approach to overcoming these inherent limitations is to use a fissile fuel in vapor, gaseous or micron-size liquid droplet phase at thousands of degrees Kelvin hotter than the materials containing the system. This permits the selection of efficient toping cycle conversion techniques such as MHD generators. This in turn requires high gas electrical conductivities, and hence ionization without excessive recombination, and fission-enhanced conductivity promises precisely such a solution. MHD energy conversion is discussed further in Section 4.5.2 and 4.5.3, under the heading of alternative nuclear power conversion cycles.

3.2.1.1 Key Features of GCR-MHD Reactor Systems. Fuel Selection Criteria for GCRs: The selection of fissile fuel and working fluid determines the properties of the partially ionized plasma exiting the vapor core outlet, and subsequently establishes the operating conditions of the energy conversion system and the closing of the cycle. Metallic uranium vapor could be considered but, because it boils at such a high temperature (5,000 K at the 10–20 atmospheres required for criticality), to keep it vaporized would pose a severe challenge for containment and separation structure materials. UF_6 has also been suggested as a lower boiling point fuel, but UF_4 proves even better. At elevated temperatures, UF_4 is thermodynamically and chemically the most stable uranium compound in vapor and liquid phase.

3.2.1.2 Generation IV Goals—Capabilities of the GCR-MHD Concept.

3.2.1.2.1 Sustainability Aspects—*Fuel Utilization*—There is no fuel burn-up limit for gas core reactors due to continuous recycling of the fuel. Due to nuclear design flexibility, a wide range of conversion ratio from completely burner to breeder is achievable. The very high efficiency and high operating temperature of gas core reactors mean that in theory lower waste per energy unit compared to the reference LWRs can be achieved, and hydrogen production using the waste heat is possible. Lower power generation cost might also be attainable although it is still too early to be able to measure the exact power generation cost.

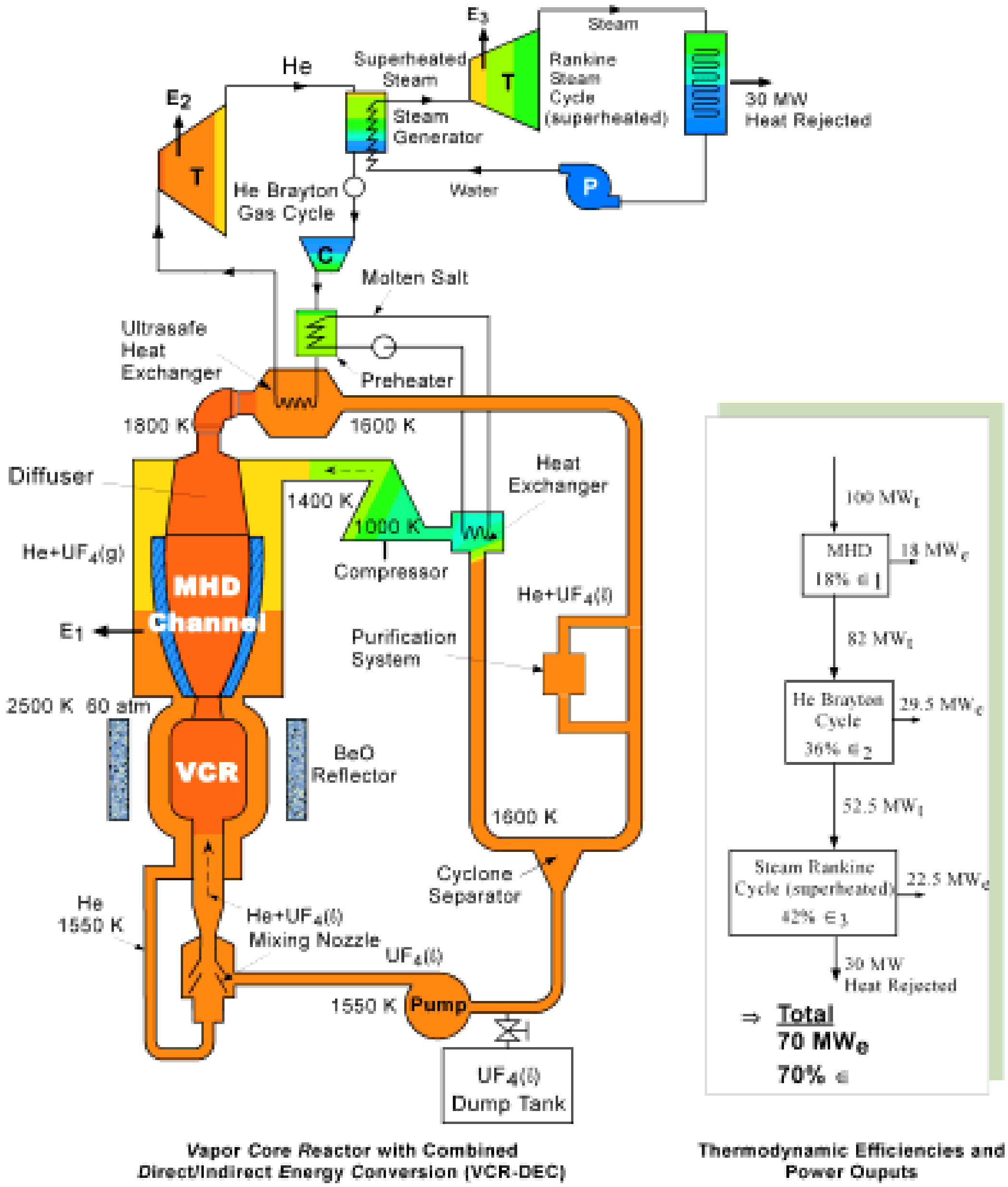


Figure 5. Schematic design for a VCR-MHD power generator with combined heat recovery cycles.

Table 6. Design characteristics of 1,000 MWe UF_4/KF Vapor Core Reactor System.

Net Electric Power	1,000 MW	Total fuel inventory	5,000 kg (UF_4 +KF)
Thermal efficiency	70%	UF_4 inventory	3,000 kg
Core volume	22.0 m^3	^{235}U	230 kg
Vessel design pressure	50 bar (735 psi)	Enrichment	10%
Average power density	65 MW m^{-3}	Total Mass flow rate	900 kg s^{-1}
Be Reflector mass	52 MT	UF_4 mass flow rate	540 kg s^{-1}

Waste Minimization—Gas core reactors feature a fully burned and integrated fuel cycle. The only waste product at the backend of the gas core reactors' fuel cycle is fission fragments that are continuously separated from the fuel. It does not require spent fuel storage or reprocessing. Due to very low fuel inventory and continuous burning and transmutation of actinides, gas core reactors minimize the environmental impact and stewardship burden.

Proliferation Resistance—Low fuel inventory (about two orders of magnitude lower than LWRs for the same power generation level) combined with continuous burning of actinides significantly reduce the proliferation threat. Even for a comparable spectral characteristic, gas core reactors produce fissile plutonium two orders of magnitude less than LWRs. In addition, the continuous recycling and burning of actinides further reduces the quality of the fissile plutonium inventory.

3.2.1.2.2 Safety and Reliability Aspects—*Worker Safety*—Due to circulation of fuel in gas core reactors, the entire operation is remote and robotically controlled. Continuous separation of fission products from fuel reduces the delayed source of radiation (after reactor shut down) to a bare minimum. These factors result in significant reduction in the probability of worker overdose during normal operation. Continuous separation of fission products may lead to an increase in accidental radiation dose to fuel processing workers.

Core Damage—Gas core reactors have better inherent safety features compared to LWRs. Neutron reflectors, that also function as external moderators, provide criticality in gas core reactors. Any loss of system pressure, core damage, or fuel leak result in loss of reactivity that is needed to keep the reactor critical. Because the fuel is in gaseous phase the core damage would be limited to pressure vessel and reflector damage that are not at the same level of severity with any other solid fuel reactor. Low thermal conductance of the vapor fuel/working fluid allows for very high average bulk fluid temperatures greater than 3,000 K, while maintaining wall temperatures significantly cooler. However, gas core reactors are at an early stage of development and the system model uncertainty is very high.

Emergency Planning—Continuous separation of fission products and low fuel inventory reduces the source term in gas core reactors by several orders of magnitude. Although no rigorous analysis exists to support the possibility of elimination of offsite evacuation, there is a good probability to design gas core reactor systems that would meet the Generation IV goal of elimination of emergency planning and offsite evacuation.

3.2.1.2.3 Economics Aspects—An evaluation of economic factors including the development, construction, and operating costs shows that a GCR-MHD power plant would incur reasonable and manageable financial and capital costs. Due to the compact nature of gas core reactor systems, the required containment building is likely to be smaller than LWRs that may lead to significant reduction in the capital cost. The cost of developing advanced structural materials that might be required for extra safety is an unknown factor. Based on the present regulatory practices, there are considerable licensing uncertainties for a reactor system that the fuel is in gaseous form and circulated. The operational temperature is much higher, and fission products are separated onsite.

Altogether, the vapor or gaseous fuel-working fluid high temperature, the broad range of operating parameters, and the potential to couple to different power conversion schemes provide the VCR with a number of attractive features, particularly regarding safety and sustainability. These are listed in Table 7.

3.2.1.3 Research and Development Challenges—GCR-MHD. The GCR-MHD concept is dependent on the development of very high temperature materials that must be able to operate reliably over long lifetimes. A related concern is the strength of materials particularly regarding protection against

Table 7. Beneficial features of Vapor Core Reactors.

Temperature and Efficiency	• Working fluid (fuel) in core can be considerably hotter than surrounding structure.
	• High working fluid T advantageous for power cycle efficiency.
Geometry	• Simple geometry and structure of core helps minimize thermal stress and thermal shock.
System Integration	• Vapor fuel-working fluid can be matched to power conversion requirements.
Fuel Utilization	• A fully integrated fuel cycle with minor actinide burning
	• Requires relatively higher uranium enrichment.
Nuclear Waste	• On line separation of fission waste that is solely included fission products.
	• No need for fuel recycling, very high actinide burning.
Safety	• Extremely low activity with minimum risk of accidental release of radiation.
	• Three orders of magnitude reduction in fuel inventory in the core—less than 1,000 kg versus more than 100 MT in comparable LWRs.[a]
	• Extremely low proliferation risk due to low fuel inventory, and actinide burning and transmutation.

a. The comparison being made here is between GCRs and Light Water Reactors (LWRs) of *comparable power output*.

the bursting of pipes. There is concern about the dependence upon fission enhancement of conductivity (how to effectively achieve this without incurring adverse radiation damage to other components, especially the electronics). Improving the vapor-fuel chemistry is another R&D challenge.

There are also many other R&D challenges that may develop after initial testing of a high temperature gas core reactor with fuel circulation. There are potential nuclear design and control issues that are not possible to reliably address without producing experimental data. There are also considerable uncertainties in the nuclear MHD power conversion process that is required to convert fission power at very high temperatures. There are also other life cycle management issues that cannot be addressed without access to reliable experimental data that does not exist.

3.2.1.4 Score Sheet Evaluation of the Gaseous or Vapor Core Reactor Concept. A previous TWG-4 meeting established a score sheet, based on polling of the TWG-4 members, for the Gas Core Reactor concept. It was intended as a screening of the concept’s potential to meet or exceed Generation IV goals. However, the concept as it stood when evaluated specifically considered a different fuel to the GCR/VCR-MHD scheme and no other gas core reactor concepts. Further research revealed three other concepts with potential to meet or exceed Generation IV reactor goals. Results of evaluation for gas core reactor concepts are presented at the end of the discussion for this concept set.

Quantitative and qualitative evaluation tables are presented in Section 3.2.5 for the grouped set of all gas core concepts as a whole.

3.2.2 Gas Core-Graphite Wall U-C-F System Concept

In all gas core reactor concepts the core wall integrity is a major cause for concern. The Gas Core-U-C-F concept attempts to solve this problem by designing the reactor core wall to be an integral participating part of the chemistry of the gas core. At the Technische Universiteit Eindhoven in the Netherlands a research group has been developing the conceptual foundations for a gas core reactor that actually uses the wall corrosion as a chemical equilibrium controlling feature.[16] To prove the viability of this concept an extensive modeling effort has been performed to determine the core conditions under which a uranium-carbon-fluorine (U-C-F) system can be kept in quasi-equilibrium with a graphite wall.

Two systems are considered, (1) a fissioning gas of U-C-F species at 1,500–3,000 K and 0.9 to 2.5 MPa, and (2) a weakly ionized fissioning gas of U-C-F-e^--ions species at 3,000–10,000 K and 2.5 MPa. A 50 MW_{th} reactor could run on a 10-year cycle with U or U-Pu recycling fuel, and with refreshment times[a] on the order of 2 to 200 hours, the layout would be as in Figure 6.

A chemical thermodynamic equilibrium computer code was used to determine the pressures and temperatures at which a gaseous mixture of UF, UF_4 and CF_4 could be kept in equilibrium with a graphite wall. The idea is to keep the dominant component UF_4 partial pressure low to avoid condensation by stimulating UF_5 formation. Calculations indicate that a fissioning gas core reactor could be operated at a pressure of 2.5 MPa with a feed consisting of a mixture of 70–93% UF_4 and 30–7% CF_4. The need to keep the wall in thermodynamic chemical equilibrium necessitates the formation of lower valent uranium and carbon fluorides, mainly CF_n, which can disrupt the core equilibrium. Above 2,300 K, UF_4 is the dominant species.

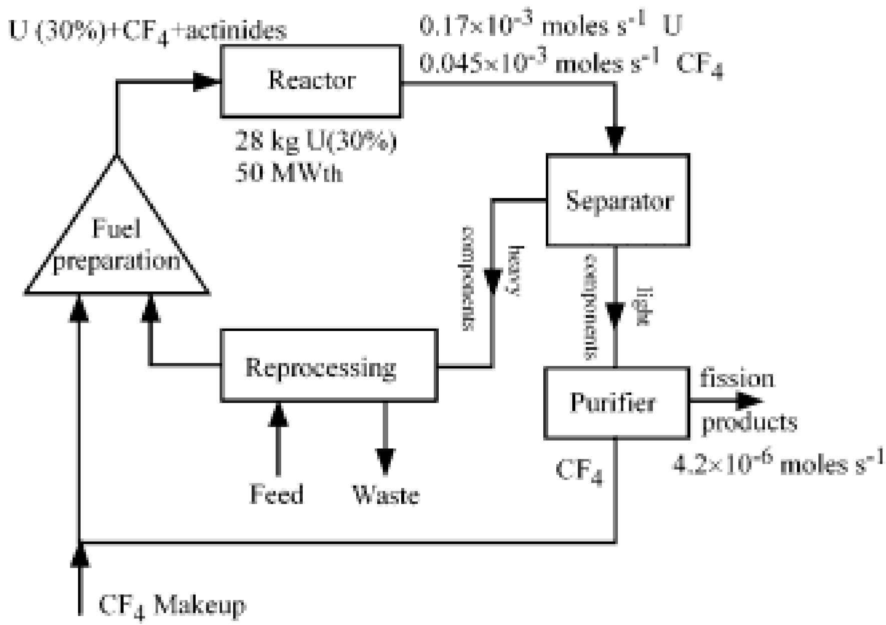


Figure 6. Quantitative block diagram of a mini-fuel cycle based on the GCR-U-F-C concept. Refreshment time = 200 h, fission product accumulation = 6.2×10^{-4} gs^{-1}. (Drawing courtesy of Technische University Eindhoven.)

a. The refreshment time is the mean residence time of the gas in the reactor, the time it takes to completely clean or replace the fuel content, but it is done continuously, so the chemical composition stays approximately constant.

The whole concept revolves around a careful analysis of the GCR wall corrosion problem. Fluorine is very corrosive, and although the aforementioned equilibrium conditions are attainable, in practice, temperature and pressure gradients disrupt the equilibrium locally, leading to serious corrosion and wall instability. Most U or C fluorides have high dissociation reaction constants at these temperatures and so free fluorine will tend to grab a lot of carbon from the wall. But by increasing the pressure or decreasing the temperature reverses the carbon balance and carbon will start to be deposited on the wall. The result is that both corrosion and deposition can occur at the wall when pressure or temperature gradients of hundreds of kPa or tens of Kelvin are present.

Various schemes have been proposed to control the corrosion or deposition at the wall. The predominant concern is stability of the reactor wall. Temperature and pressure gradients in the direction of flow parallel to the wall will imply changing chemical equilibrium conditions, meaning that ultimately a fixed equilibrium gas composition cannot be maintained. To maintain stability some form of control or feedback is needed at the wall to ensure that the relatively thin pyrolitic graphite tiles being used do not wear away at localized spots. This would expose the outer ceramic and niobium steel vessel support structure to heat and chemical attack.

Considerations of neutron economy in a GCR-U-C-F concept focus on removing neutron-hungry fission products. But over a life cycle of 5 years, with medium enrichment (30% ^{235}U), neutron absorption is considered to be only a minor nuisance. A more serious concern is the build-up of fission product fluorides, with the main culprit being SrF_2. The added partial pressures of fission product fluorides degrade the GCR performance and core regulation. They can also be challenging to remove because they condense on pipes unless the walls can be kept higher than 1,800 K and gas flow kept very high at 2.5×10^{-4} m^3s^{-1}.

For the work producing part of the U-C-F-e^--ion cycle, the Dutch team envisage a magneto-inductive piston, providing pulsed power driven by an LC-circuit. A typical magnetic piston generator would compress the fissioning gas to 10 MPa and 10,000 K, with a magnetic field of 5 Tesla, generating an extra 0.8 MJ m^{-3} additional fission energy release. The cycle frequency would be 50 Hz. This is possibly not as efficient as MHD (see above and Section 4.5.3) but has the advantage of being cleaner than MHD generation because with magneto-inductive pistons there is no need for solid-state electrodes in contact with the hot partial plasma.

3.2.2.1 Generation IV Goals—Capabilities of the GCR-U-C-F Concept. There is a commonality of design between all gas core reactor concepts. In that regard, they share many sustainability, safety, and economy attributes that are pertinent to the Generation IV reactor goals. Following are brief comments regarding unique features of the GCR-U-C-F concept.

3.2.2.1.1 Sustainability Aspects—Fuel utilization, waste minimization, and proliferation resistant capabilities of GCR-U-C-F concept are very similar to the GCR-MHD concept. Recycling is simplified and refueling is continuous. Fissile material in the gas phase reduces the fissile inventory by orders of magnitude compared to classical solid core reactors.

3.2.2.1.2 Safety and Reliability Aspects—Wall integrity is not necessarily a risk factor because wall corrosion will lead to increased cooling with a consequent negative feedback. The high temperatures in the core are no longer considered a threat to the vessel because the core chemistry actually relies upon the wall corrosion. For the same reason the nuclear chain reaction is not suppressed to the same extent that it must be in conventional nuclear reactors. Core damage is a non issue because there is no possibility of melting.

3.2.2.1.3 Economic Aspects—Fissile material in the gas phase reduces the fissile inventory by orders of magnitude compared to classical solid core reactors. The nuclear fission energy is not degraded to heat, but is used directly for electricity production via magneto-inductive motors (magnetic pistons). The exact lifecycle costs are little understood, but supposing a level playing field, given the necessary R&D goals can be met, then the GCR-U-C-F concept should come out faring much better than the reference case.

3.2.2.2 Research and Development Challenges—GCR-U-C-F. The equilibrium chemical compositions can be calculated but they unfortunately turn out to be very sensitive to the thermodynamic property data that is used. A small discrepancy of say 5 to 10% in the free energy of formations may result in significantly different equilibrium compositions or otherwise significant change in operation temperature. For example, in the late 1970s the calculations showed no significant amounts of UF_4 were produced in equilibrium, but since then, new data published on uranium fluorides have been incorporated (with lower values for the Gibbs free energies of UF_5 and UF_6) and they result in thermodynamic models that do produce significant UF_5, in fact UF_5 becomes the dominant component in the temperature range from 1,800–2,300 K. Also, the presence of moisture, H_2O, did not appear to be included in the chemical equilibrium calculations. So although HF is sure to be present in the U-C-F system it is not clear how severely this will complicate matters.

Control systems of some complexity are required to limit the damage due to corrosion and deposition of carbon at the GCR walls that will inevitably occur due to localized temperature and pressure gradients in a continuous gas flow. In general, a loss of cooling over a certain area of the wall will increase the corrosion at the same place. A favored localized-corrosion avoidance method is to use a "Stepped Flow" device, whereby sudden changes in the *P* and *T* are made which avoid the corrosion and deposition effects that would occur along the walls in a continuous expanding-contracting flow at constant *T*. But this requires a technology to be developed, such as some kind of membrane to provide a sharp step in the *P* and *T*. Another possibility is that the pyrolitic graphite tiles could be connected to the ceramic body like the heat shields are on the space shuttle to provide maximum mechanical contact. Localized wall corrosion should then increase heat transfer and thus lead to increased cooling and therefore a negative feedback leading to cessation of further corrosion. Alternatively the wall integrity needs to be monitored to ensure safe operation over the projected lifecycle.

Fuel composition and removal of fission product fluorides is a concern, either short refreshment times are required or some technological hurdles overcome in order to avoid unwanted pressure build-up and fouling due to fission product fluorides condensing in valves and such. Volatile radioactive species leakage is another problem shared with all gas core concepts.

In this design, without the nuclear enhancement of conductivity, the working leg of the cycle requires high temperatures ($T>6{,}000$ K) in order to achieve the gas conductivity that would make magnetic-induction pistons efficient.

3.2.3 Mixed Vortex Flow UF_6 Gas Core Reactor Concept

Los Alamos National Laboratory has prepared preliminary analyses for gas core reactors of two types. One is a mixed flow UF_6 concept; the other is a plasma core reactor with radiative heat transfer concept. This section describes the mixed flow GCR concept (GCR-UF6).

3.2.3.1 Overview of GCR-UF6 Reactors. The GCR-UF6 reactor type was proposed in an effort by Los Alamos Scientific Laboratory (LASL) to meet U.S. Energy Research and Development Administration official's demands for nuclear power plants capable of achieving:

- Low fissile material inventory

- Low fissile material divertability
- Minimal fissile material transportation.

In other words the demand was for reactors having low proliferation potential. A further criterion was potential for use in developing countries, which was translated to mean that low power outputs would be quite acceptable. Two similar Mixed Vortex flow designs were analyzed, one a 200 MW_{th} compact reactor, the other a 2,500 MW_{th}. Both have molten salt breeder blankets for fuel recycling. Power extraction is via heat exchangers external to the main reactor where the vortex heats up the fissioning fuel.

Mixed flow refers to the working fluid composition of UF_6 and Helium. Helium is chosen for its heat transfer properties. The vortex flow refers to the method of wall protection against the high temperature corrosive gas constituents. Figure 7 shows a cut-away of the whole system consisting of seven individual inner vortex flow tubes. Establishing a corkscrew-like vortex flow in the reactor vortex flow tubes means that a fuel gas centerline temperature of 1,500 K can be maintained while the temperature at the wall is only 500 K. The vortex flow is produced by tangentially injected fluid, and most of the fluid actually exits the tube through perforations in the side of the tube. This is all part of the design to keep the wall temperature well below the point where UF_6 dissociates significantly (requiring T<2,000 K). Materials considerations (particularly corrosion resistance investigations) and neutronics analyses have reasonably rigorously proven that the GCR-UF6 Vortex Flow concepts can meet the Generation IV feasibility goals,[17,18] if not the full set of goals.

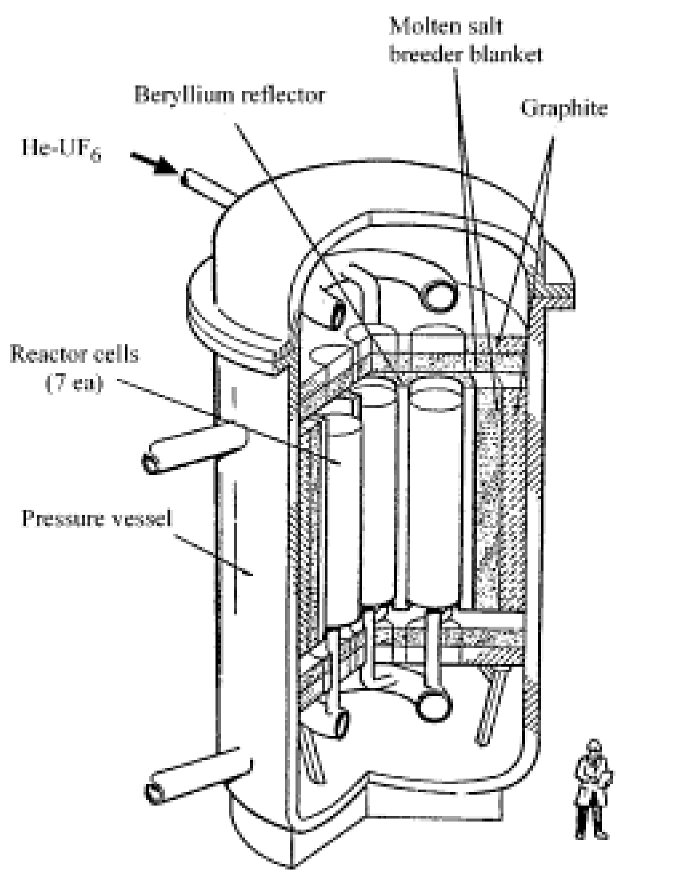


Figure 7. UF_6 Gas Core Reactor Vortex Flow system configuration. (Original drawing provided by LANL.)

The gas temperature is limited to about 2,000 K to avoid any significant UF_6 dissociation. The 200 MW_{th} plant has a cavity pressure of 100 atm, and the individual vortex flow tubes are 1 meter in diameter and 4.35 meter long. The mass flow rate is 68 kgs^{-1}, the critical uranium mass in the cavity is 45 kg, and the total inventory is 91 kg.

The 2,500 MW_{th} plant needs a high pressure of 650 atm, and its vortex tubes are 8.5 meter long. The mass flow rate is 625 kgs^{-1}, the critical uranium mass in the cavity is 45 kg, and the total inventory is 191 kg.

3.2.3.2 Generation IV Goals—Capabilities of the GCR-UF6 Concept. There is a commonality of design between all gas core reactor concepts. In that regard they share many sustainability, safety, and economy attributes that are pertinent to the Generation IV reactor goals. Following are brief comments regarding features of the GCR-UF6 concept.

3.2.3.2.1 Sustainability Aspects—*Fuel Utilization*—On-site continuous fuel recycling experiments have shown early promising results. No fuel rods need to be manufactured or recycled.

Waste Minimization—GCRs are intrinsically efficient fuel users due to the gaseous phase burning and high temperatures. Online fission product removal will facilitate waste handling.

Proliferation Resistance—With breeding ratios of ≈1 the only chance for fissile material hijacking is the initial mass of 191 kg (for a 2,500 MW_{th} mixed flow design). Less than 4 kg could be surreptitiously diverted before the reactor would shut down. Since the entire inventory is less than one-tenth of current nuclear power plants, the overall proliferation risk of GCR-UF6 reactors is extremely low.

3.2.3.2.2 Safety and Reliability Aspects—The concept cannot be rated yet as excelling in safety and reliability, but some steps towards the goals can be identified.

Worker Safety—Fluid mechanical vortex confinement of UF_6 at high enough densities to sustain criticality has been experimentally demonstrated. Fuel injection and buffer gas flow control techniques have been investigated.

Core Damage—The Core Damage accident scenario is eliminated as in all gaseous or vapor core concepts. Leakage and vessel integrity attack by corrosive chemical species are, however, a big worry. Nickel or Ni-Al alloy liners are thought to be the best corrosion resistant materials, but nickel also has a high neutron capture cross section, so only a thin liner or cladding is feasible.[17] One fortunate property is that the thermal expansion coefficients of Ni and Be are almost identical, which raises the possibility of some kind of composite containment vessel structure.

Emergency Planning—Although leakage of fission products is a concern, in the event of an emergency the offsite response would be minimal because the fuel automatically condenses, and can be easily contained by adequate containment structures within current capabilities.

3.2.3.2.3 Economic Aspects—*Capital and Production Costs*—The lifecycle cost advantage is at least comparable with the reference cases. The high temperature operation should ensure efficient energy production when all other factors are considered equal with the reference LWR. The reactor does not need to be shut down for recharging. Exact estimates of lifetime costs are lacking.

Profitability—Safety considerations will incur costs but the financial risks are low because the design can be made safe with current confinement and safety technologies.

3.2.3.3 Research and Development Challenges—GCR-UF6. High temperature uranium fluorine chemistry is a source of uncertainty. The corrosive nature of fluorine necessitates a wide search for suitable wall lining materials that will not hinder the neutronics. While Ni-Al has proven superior resistance to F attack, Ni-Mg alloys should also have good corrosion resistance but have not been investigated. If either Ni-Al or Ni-Mg prove to be too unfavorable to the neutronics, then a completely different tack would be to consider even more corrosion resistant materials like Au or CaF_2, but the higher costs of fabricating such corrosion resistant materials would have to be evaluated. In summary, R&D needs are (2) accurate corrosion rate data for a variety of possible structural materials in realistic fluorine environments likely to be encountered, and (2) realistic simulations of core operating conditions employing the corrosion rate data.

If the original goal of a compact, highly safe reactor for use in developing countries is to be achieved, then a careful evaluation of the safety aspects needs to be made, including waste product removal and handling, as well as a definition of the remote maintenance requirements.

3.2.4 Plasma Vortex Flow Gas Core Reactor Concept

This section describes the plasma core vortex flow GCR concept. Two different types of plasma core systems are considered for Generation IV potential. The main reference for this concept summary comes from Los Alamos National Laboratory where the ideas originated and were initially conceived as solutions specifically for low proliferation risk nuclear power.[19]

3.2.4.1 Overview of Plasma Vortex Reactors. One type of plasma vortex gas core reactor runs a closed cycle helium gas turbine, but uses a metallic uranium vapor with argon gas as the plasma fuel. The fissile inventory in total in the cavity is 36 kg, and the total plant inventory is 65 kg.

The second plasma vortex reactor concept is a "Transmission Cell" plasma reactor, in which radiative energy transfer is exploited to heat the working fluid that flows around an outer channel. These are also metallic uranium-argon vapor plasma core reactors. The motivation behind this high radiant energy flux design is specifically the intention to use this type of plasma reactor in high performance special needs power applications such as (1) high thrust, high specific impulse nuclear electric propulsion, (2) photochemical or thermochemical process industry application, specifically hydrogen production, and (3) MHD power conversion for electricity production. This reactor's critical mass is even lower still, with only 16 kg in the cavity, and the total plant inventory is 45 kg.

As with the mixed vortex flow concepts, the hot (now T>5,000 K) gas is held away from the core wall by vortex buffer confinement. Both plasma vortex designs were analyzed with similar neutronics and moderator characteristics, and with the same molten salt breeder blankets. The cavities were 0.5 meter in diameter and 1.85 meters in length. Both concepts have cavity pressures of 500 atm, and mass flow rates of 237 kgs^{-1}. They only differ in their fissile inventory and method of energy extraction and associated cell structure.

3.2.4.2 Generation IV Goals—Capabilities of the Plasma Vortex Concept. Following are brief comments regarding unique features of the Plasma Vortex gas core reactor concept, in terms of sustainability, safety, and economics.

3.2.4.2.1 Sustainability Aspects—*Fuel Utilization*—No fuel rods need to be manufactured or recycled. Extremely low inventory and online recycling extend the life of fuel.

Waste Minimization—Plasma cores have very high efficiency fuel utilization.

Proliferation Resistance—With breeding ratios of ≈1 the only chance for fissile material hijacking is the initial mass of 64 kg (for the He turbine cycle design). The entire inventory is hundreds of times smaller than that of current conventional nuclear power plants, making the overall proliferation resistance of these plasma core reactors superior to other designs.

3.2.4.2.2 Safety and Reliability Aspects—*Worker Safety*—Fluid mechanical vortex confinement of the high temperature gas should protect workers. The main challenge would be in getting sufficient stability for power production. This is a key feasibility issue that may prove to be the showstopper for this concept.

Core Damage—The Core Damage Accident scenario is eliminated as in all gaseous or vapor core concepts. Loss of control will usually mean at worst some equipment damage, but as the fuel is gaseous there is no fuel damage, and so rapid restarts are at least possible in the majority of accident scenarios.

Emergency Planning—Although leakage of fission products is a concern, in the event of an emergency the off-site response would be minimal because the uranium vapor will condense, and can be easily contained by adequate containment structures within current capabilities.

3.2.4.2.3 Economic Aspects—*Capital and Production Costs*—No shut down for recharging, online fuel recycling and efficiencies approaching 70% are possible, leading to potentially very attractive lifecycle economies.

Profitability—The plasma vortex concept is simple enough to warrant financial risk provided the fluid dynamic flow containment issue could be solved.

3.2.4.3 Research and Development Challenges—Plasma Vortex. High temperature uranium fluorine chemistry is a source of uncertainty. The corrosive nature of fluorine necessitates perhaps a wider search for suitable wall lining materials that will not hinder the neutronics. Fluid dynamic models need to be worked out for the effective confinement of the fuel mixtures. In particular, the dependence on achieving a vortex flow introduces questions of whether flow instabilities could hinder reactor control, and so the sensitivity of the reactor control variables to variations in flow characteristics become crucial R&D goals. Failure of primary loop piping and of the pressure vessel need to be simulated so that the ramifications for safety and shutdown times can be understood. Remote maintenance requirements must also be carefully analyzed.

3.2.5 Concept Viability Evaluation of Gas Core Reactors

The score sheet developed through a detailed concept presentation and discussion within the Nonclassical Concept TWG is presented in Table 8. A previous TWG-4 meeting established the score sheet, based upon consensus. It is intended as a qualitative screening of the concept's potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

Evaluation of the potential to meet or exceed Generation IV goals was completed by the TWG-4 for all of the gas core reactor concepts considered as a whole group, yet with characteristics of each individual concept factored in to the scores nonetheless. Scoring for screening potential of the GCR/VCR-MHD concept was based on the most viable gas core reactor design that uses chemically stable uranium tetrafluoride fuel with MHD generator as an integral part of the design. That concept uses Brayton and Rankine bottoming cycles to increase conversion efficiency.

Table 8. Qualitative Group Summary Evaluation—Gas Core Set.

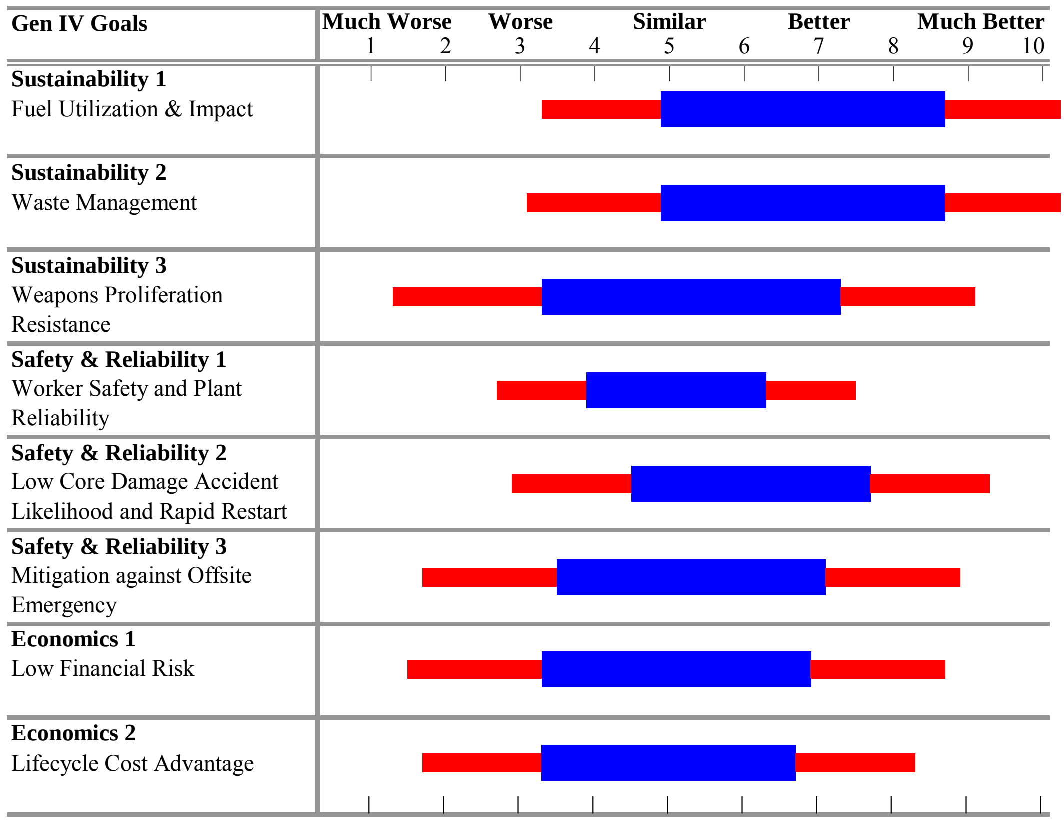
Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization and Impact	++	• Ultrahigh burn-up, ultrahigh temperature • Low fuel inventory, high conversion ratio
Sustainability 2 Waste Management	++	• Low activity • Low waste efflux • Online waste removal
Sustainability 3 Weapons Proliferation Resistance	+	• Near perfect actinide burner • Superior to most other reactor concepts
Safety and Reliability 1 Worker Safety and Plant Reliability	+	• Fission products are continuously removed • Few moving parts • Negative reactivity coefficient
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	+	• Immediate loss of criticality upon CDA • Complete elimination of fuel damage
Safety and Reliability 3 Mitigation against Offsite Emergency	+	• Ultra-low radioactive inventory • If leaks, gaseous fuel condenses • Extremely low source term
Economics 1 Life-Cycle Cost Advantage	+	• Excellent conversion efficiency • Hydrogen energy production from waste heat.) • Space applications likely
Economics 2 Low Financial Risk	=	• Comparatively simple design • Compact containment building • High development cost

The decision to retain the GCR concepts for further Generation IV evaluation was arrived at after considerable deliberation by members of TWG-4, and fully takes into account all the strengths and weaknesses of the gas/vapor core reactor concepts.

In addition, results of independent polling/evaluation of the members the TWG-4 are presented in Table 9. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean. Note that the mean scores in this table are close to the respective qualitative scores of for the respective lumped goal categories.

Note that the wide spread of scores can be attributed both to the uncertainty of the data, low level of technology maturity of the gas core reactors, and the fact that all four gas core reactor concepts were evaluated collectively, with the single scores under each category representative of the combined set of four concepts. The spread of scores indicated in this table also reflects both the uncertainty in the available data, upon which the 14 TWG-4 members had to base their assessments, as well as the expanded scale that now ranges from 1–10, rather than the five-fold (– –,–,=,+,++) scale used in the qualitative consensus-based screening. Under each goal there were a number of criteria and metrics, all of which were scored, but only the category averages are presented in Table 9. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean.

Table 9. Quantitative Score Sheet Evaluation for the Gas Core Reactor Group.



3.3 Nonconventional Coolant Reactors

Number	Concept Name	Sponsorship
N6	AHTR—Molten Salt Cooled, Graphite Matrix	ORNL & SNL
–	OCR—Organic Cooled Reactor	AECL/U. of Wisconsin
–	FSEGT—Sodium Evaporation Cooled, Gas Turbine	IPPE, Russia

Nonconventional Coolant Reactors use molten salts or high boiling point organic liquids to remove core power at lower pressures and provide heat at higher temperatures than conventional comparable coolants. A reactor design based on a molten salt coolant allows low-pressure systems with exit temperatures above 1,000°C. This system can be used to produce hydrogen from water or to generate electricity at greater than 50% efficiency using an indirect power cycle. Organic coolant compounds feature high concentration of hydrogen and low vapor pressure, and are meant to replace the moderating power and enhance the heat transport capability of water at lower system pressures. Metallic vapor concepts represent another potential nonconventional coolant approach that could also fit in to the Metal or Nonclassical categories. The two concepts described below are a molten-salt-cooled, graphite-matrix-fueled reactor for either hydrogen or very high efficiency electricity production, and an organic cooled system based on Atomic Energy of Canada, Ltd. (AECL) heavy water cooled natural uranium reactor (CANDU) technology that allows for operation at lower pressure and more convenient maintenance.

3.3.1 Modular, Molten-Salt-Cooled, Graphite-Matrix-Fuel Advanced High Temperature Reactor

3.3.1.1 General Design Characteristics. The modular, molten-salt-cooled, graphite-matrix-fuel, advanced high temperature reactor (AHTR) is designed to provide heat at the conditions (high temperatures and low pressures) necessary to create new nuclear energy options.[20] These options are: (1) hydrogen production by thermochemical water splitting, which requires 800 to 1,000°C heat, and (2) advanced electric production methods (indirect gas turbine cycles and direct thermal to electric techniques). AHTR systems could be designed as loop or pool type systems from a few hundred MW_{th} to more than 2,000 MW_{th} with outlet temperatures >1,000°C. The fuel is a coated particle, graphite-matrix fuel with the same general characteristics as the fuel developed for high-temperature gas cooled reactors (HTGRs). These coated particle fuels operate at temperatures in the 1,200°C range with shorter term transient capabilities to near 1,600°C. The AHTR fuel cycle would be essentially similar to the HTGR. The coolant would be a molten fluoride salt ($2LiF-BeF_2$) developed for molten-salt-fueled fission reactors and cooling the first wall of fusion reactors. The molten salt coolant and graphite moderator matrix are compatible at temperatures of 1,000°C. The molten salt has an atmospheric boiling point of ~1,400°C. The excellent heat transfer properties of the molten salt, compared to helium, reduce the temperature drops between the fuel and molten salt and the molten salt and any secondary system. Other molten salt compositions with differing thermal and neutronic property trade-offs can also be considered for reactor coolant applications. These characteristics allow designs with higher coolant temperatures than in gas-cooled systems for the same maximum temperature limit in the fuel.

The reactor core consists of graphite-matrix fuel cooled with a molten salt. Moderation in the graphite moderator and molten salt coolant results in a thermal neutron spectrum within the reactor core. The primary heat transfer loop consists of the reactor core, primary heat exchangers and salt pumps for forced circulation of the hot molten salt (~1,000°C). Natural circulation of the molten salt provides an effective mechanism for passive decay heat removal. A schematic layout is shown in Figure 8.

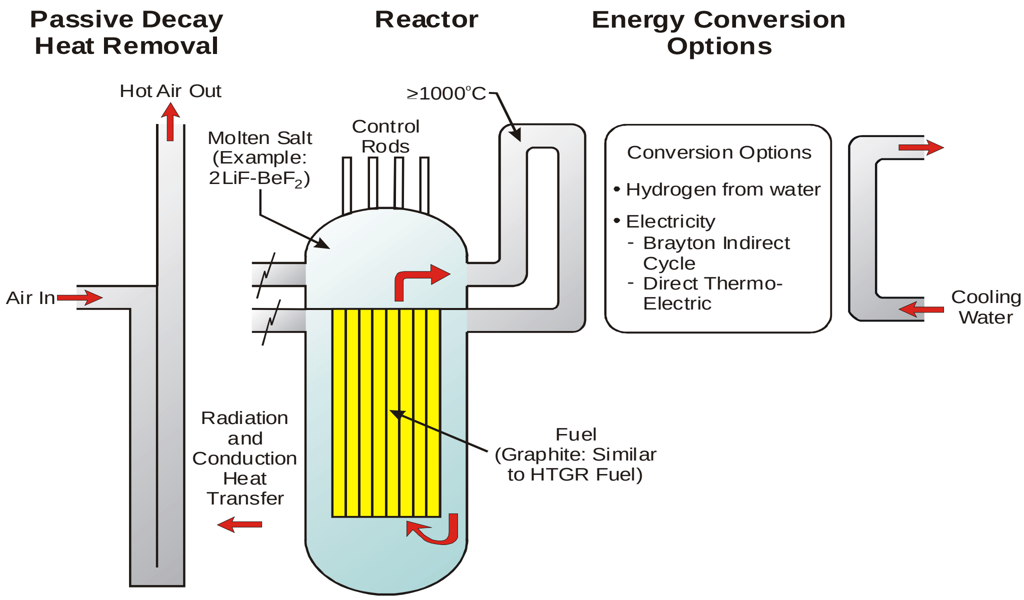


Figure 8. Schematic of advanced high-temperature reactor (AHTR). (Drawing courtesy of ORNL.)

3.3.1.2 AHTR Applications. The AHTR makes use of existing fuel and coolant technologies to produce a very high outlet temperature of ~1,000°C to enable, (1) efficient thermochemical production of hydrogen, or (2) very high efficiency electricity production, using an indirect Brayton cycle or in the future possibly direct production of electricity from heat. These very high outlet coolant temperatures are made possible by the compatibility of the molten salt and the graphite moderator matrix at these elevated temperatures, and the inherent high temperature fission product retention of the coated particle fuel at temperatures up to 1,200°C.[21,22] The same basic coated-particle fuel used for helium-cooled graphite-fuel reactors is the baseline fuel used in the AHTR. The transfer of heat from the primary to secondary system will depend upon the application. A secondary heat transfer loop would be used if the reactor were coupled to an indirect helium Brayton cycle. The secondary heat transfer loop isolates the low pressure primary system from the high-pressure Brayton cycle.

For hydrogen production, a secondary heat transfer loop or a direct radiation heat exchanger may be used. Direct radiation heat transfer involves tubes with molten salt radiating to tubes containing chemical reagents used for thermochemical hydrogen production. Designs for high temperature heat exchangers based on radiative heat transfer at these elevated temperatures appear to be practical. The tubes in any one sheet may be connected with welded metal plates between the tubes to increase the radiative heat transfer surface and improve radiation coupling. Such sheets of tubes would be almost identical in design to the boiler-tube curtain walls used in fossil boilers. This method would provide improved isolation between the molten salt and highly corrosive chemicals in thermochemical production systems.

3.3.1.3 Generation IV Goals—Capabilities of the AHTR Concept.

3.3.1.3.1 Sustainability Aspects—*Sustainability 1*—The fuel cycle options are essentially identical to those of other graphite based gas cooled systems. These include various low-enriched uranium and low-enriched uranium thorium fuel cycles that could be chosen to fall within nonproliferation guidelines. The uranium and thorium consumption would be less than an LWR because of the higher neutron efficiency, and the higher overall plant efficiency, potentially 50 to 55% for electrical conversion.

Sustainability 2—The graphite fuel matrix is an outstanding waste handling form in terms of repository performance and thus reduces long-term stewardship burdens.

Sustainability 3—The proliferation resistance is superior to an LWR because there is less plutonium in the spent nuclear fuel per unit energy produced and the plutonium isotopics are shifted toward lower ^{239}Pu concentrations.

3.3.1.3.2 Safety and Reliability Aspects—*Safety and Reliability 1*—Worker safety would be expected to be similar to a LWR. The low power density and high heat capacity and passive heat transfer capability of the molten salt coolant significantly reduce or eliminate many major accident scenarios.

Safety and Reliability 2—The low-pressure primary system coolant reduces the need for high-strength materials in the vessel and external heat exchangers, compared to other high-pressure fluid systems. The AHTR design allows neutronic and thermal passive safety approaches to be used. The high-temperature Doppler effect within the thermal spectrum fuel limits reactor power. Because the molten salt expands upon heating, an additional negative moderator temperature coefficient is associated with coolant heating and expansion.

Safety and Reliability 3—In a loss of coolant accident, the decay heat is conducted directly from the reactor core through the steel pressure vessel, and to the environment via cooling ducts near the reactor vessel. The reactor is assumed to be depressurized in the case of a helium-cooled reactor accident, which minimizes heat transfer by convection. For the AHTR, if it is assumed that the molten salt has leaked from the reactor system, the reactor core heat transfer conditions would be essentially identical to gas cooled systems. For pool type designs, however, where the loss of coolant may be considered less probable, the potential for significant heat transfer from molten salt convection significantly improves passive safety margins. The liquid coolant lowers the potential for radionuclide release by several mechanisms. One is low pressure, which reduces the driving force for radionuclide transport in an accident from the reactor core to the environment due to entrainment in high-pressure gases leaving the system. The absence of pressurized fluids not only eliminates the primary driving force for radionuclide release- but also reduces containment or confinement system structural requirements and simplifies isolation of the reactor from the environment. Another mechanism is natural circulation. The large heat capacity and density allow natural circulation of the coolant to provide decay heat cooling. Most fission products and actinides are highly soluble in the salt, which provides an additional impeding mechanism or hold-up in the release of fission products from the salt mixture in the case of a breach of the reactor vessel or primary piping.

3.3.1.3.3 Economic Aspects. Hydrogen production by electrolysis is relatively efficient (~80%). However, when it is combined with the electrical conversion efficiency ranging from approximately 34% in current LWRs to 50 % for advanced systems, the overall efficiency would be in the range of 25 to 40%. For thermochemical approaches, an overall efficiency of >50% has been projected (refer also to the discussions in Section 4.2, and in Appendix B). These higher efficiency thermochemical cycles however require production system inlet temperatures in the range of 800 to 1,000°C. Due to the improved heat transfer and thus lower temperature drops between the fuel and coolant secondary systems for the liquid salt (compared with that for helium gas), the maximum salt outlet temperature can be significantly higher than that for a gas cooled reactor with the same graphite fuel and same peak fuel temperature limits. The low pressure primary coolant conditions also match the low pressure conditions across the heat exchanger in a thermochemical hydrogen production facility, making the molten salt cooled system ideally suited for high temperature hydrogen production.

3.3.1.4 Research and Development Challenges—AHTR. Considering feasibility, three advantages of the AHTR include:

1. High temperature outlet that directly addresses the requirements for hydrogen production, high efficiency electricity, or other applications.

2. Inherent safety characteristics based on low power density, high heat capacity and efficient heat transfer.

3. Reduced R&D requirements due to the use of existing fuel and coolant technology to reduce the R&D requirements.

However, meeting these goals implies operating at high temperatures, which presents serious engineering challenges. Four major needs have been identified:

- *Materials* (primary heat exchangers, high temperature molten salt compatibility with structural, vessel, other metals).

- *System design* (complex tradeoffs for high temperature systems, minimum temperature drop fuel element configurations).

- *Heat exchangers* (radiative heat transfer designs).
- *Hydrogen production* (development work to optimize and tailor thermochemical hydrogen production cycle for nuclear heat source).

3.3.2 Concept Viability Evaluation for the AHTR Concept

The score sheet developed by consensus within the Nonclassical Concept TWG is presented in Table 10. A previous TWG-4 meeting established the score sheet, based upon consensus. It is intended as a qualitative screening of the concept's potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

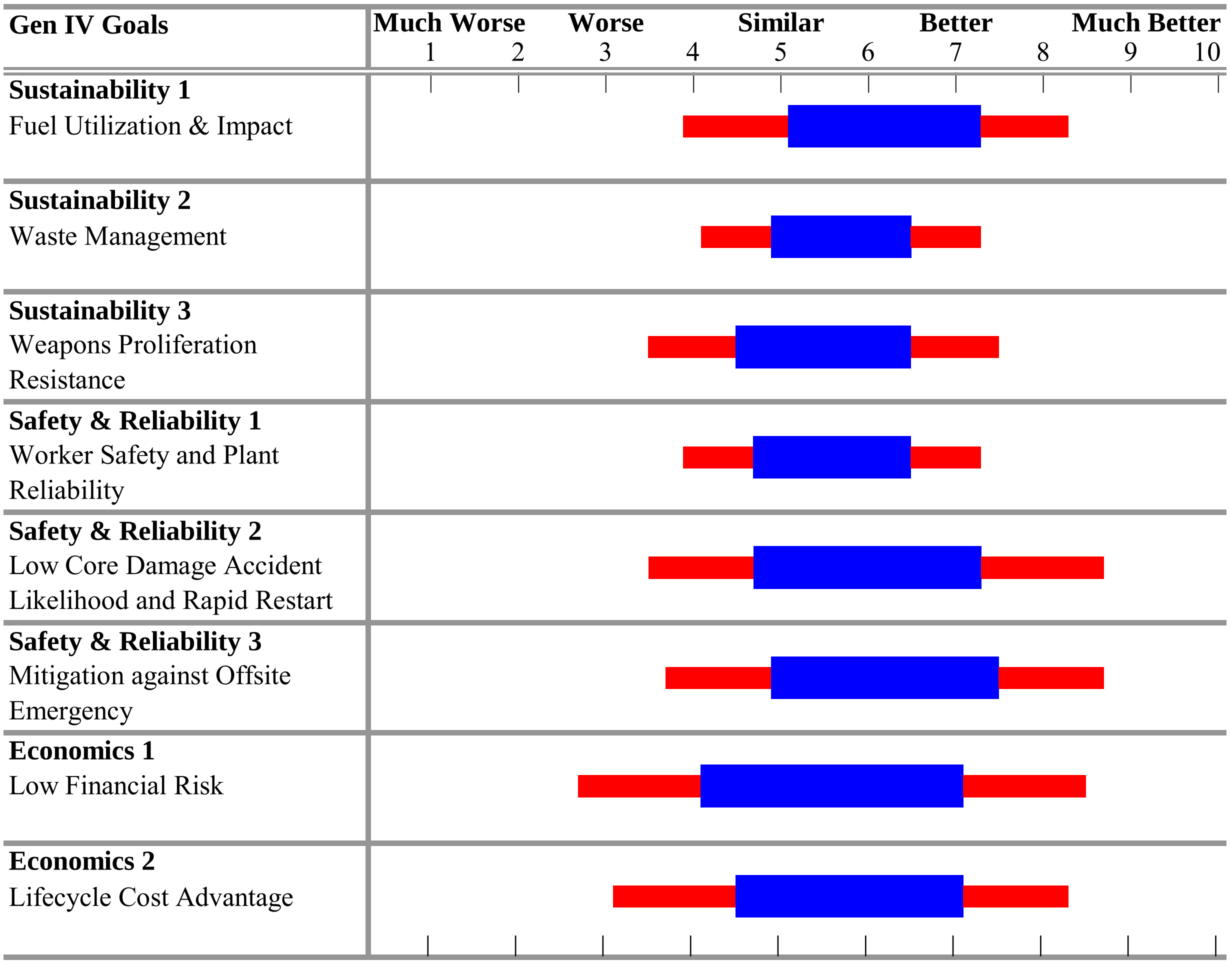
The decision of the TWG-4 at this time is to retain the AHTR concept for further Generation IV evaluation.

An additional, more quantitative, summary score sheet of the scoring for potential to meet or exceed Generation IV goals is presented for the Modular AHTR concept in Table 11. To compile this table, the members of the TWG-4 were independently polled on all the categories defined by the Generation IV evaluation and methodology group. Under each goal there were a number of criteria and metrics, all of which were scored, but only the category averages are presented in Table 11. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean.

Table 10. Qualitative Summary Evaluation—AHTR concept.

Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	+	• High neutron efficiency • High plant efficiency
Sustainability 2 Waste Management	+	• Appropriate waste form
Sustainability 3 Weapons Proliferation Resistance	=	• Similar to gas cooled
Safety and Reliability 1 Worker Safety and Plant Reliability	+	• Low pressure operation • Transparent fluid system
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	+	• Low power density • Passive decay heat removal • High heat capacity
Safety and Reliability 3 Mitigation against Offsite Emergency	+	• Passive heat removal • Low pressure system (no driving force) • FP retention
Economics 1 Life-Cycle Cost Advantage	+	• High thermal efficiency
Economics 2 Low Financial Risk	+	• Multiple energy products • Potentially simple vessel, containment design

Table 11. Quantitative score sheet evaluation of the AHTR concept.



The large spread of scores indicated in this table reflects both the uncertainty in the data that was available for the 14 TWG-4 members to base their assessments upon, as well as the expanded scale that now ranges from 1–10, rather than the five-fold scale (– –,–,=,+,++) used in the qualitative consensus-based screening.

3.3.2.1 Summary Evaluation for AHTR. *Sustainability-1*—AHTR – better than the reference due very high efficiency and high conversion fuels cycle options.

Sustainability-2—AHTR – better than the reference due to high thermal efficiency (less waste produced per MWhr)

Sustainability-3—AHTR – similar to the reference, potential for higher burn-up, but also higher conversion.

Safety & Reliability-1—AHTR – better than the reference because of low-pressure primary coolant system and low activation of primary coolant.

Safety & Reliability-2—AHTR – determined to be “better” by the working group, but rated as “much better” than the reference by the advocate subgroup due to inherent safety features, low power density, high heat capacity, efficient heat transport to boundaries for passive decay heat removal. Negative fuel and void reactivity coefficients

Safety & Reliability-3—AHTR – determined to be "better" by the working group, but rated as "much better" than the reference by the advocate subgroup due to fission product retention in molten salt, high temperature fission product retention in coated particle fuel, low-pressure system, passive decay heat removal.

Economics-1—AHTR – better than reference due to high thermal efficiency, low-pressure vessel and components. Limited experience base for estimating construction costs.

Economics-2—AHTR – determined to be similar to the reference by the advocate subgroup, however, the working group score average indicated slightly better than the reference, indicative of perceptions that the engineering feats would be achievable with modest expenditure.

3.3.3 Organic Cooled Reactors (OCR)

In addition to early interest by Italy and the United States, the most complete development of an organic cooled reactor is the CANDU-Organic Cooled Reactor (OCR) conceived by Atomic Energy of Canada, Limited (AECL), as a modification of their CANDU Pressurized Heavy Water Reactor (PHWR).[23] This reactor concept was originally developed in the mid 1970s and shelved due to a business decision and not due to a technical shortcoming. AECL operated a 60 MW_{th} organic cooled research reactor, WR-1, for 20 years that provides a body of experience with such reactor systems.

The major differences between a CANDU-PHWR and the CANDU-OCR are:

- Replacement of the heavy water coolant within the pressure tubes with organic coolant (HB-40)
- Zr-clad uranium-carbide fuel bundles
- Vertical orientation with single fueling machine.

The noteworthy features of a CANDU-PHWR that are retained include:

- Continuous online refueling
- Ability to operate with natural uranium fuel and other high conversion fuel cycles
- Low temperature heavy water moderator.

3.3.3.1.1 Advantages of OCRs over Conventional Coolant Reactors—The organic coolant provides three basic advantages: lower pressure/higher temperature operation, lower coolant activity, and lower cost coolant compared to standard CANDU reactor. The primary coolant outlet conditions in a 500 MWe reference design would be 1.4 MPa (~200 psi) at 400°C. This leads to a number of secondary benefits including higher thermal efficiency, lower voiding rate in loss-of-coolant transients and decreased capital cost. The organic coolant is much less corrosive than water and will have a much lower activity. This leads to a significant secondary benefit of lower operation and maintenance costs because of increased access to the primary system. During the operation of the WR-1 reactor, the primary heat exchanger and pump rooms were accessible during full power operation. By replacing the heavy water coolant with organic coolant, the tritium production and potential emission would be much lower.

The UC fuel has both improved thermal properties and fission product retention behavior. The higher thermal conductivity allows the UC fuel to operate at a lower temperature and the higher melting temperature provides greater margins in transient scenarios. The improved retention of fission products has an impact on reducing the source term in certain accident scenarios.

3.3.3.1.2 Disadvantages of OCRs—The organic coolant has two disadvantages: fouling/coking of heat transport system and flammability. There is concern that in low flow regions of the primary heat transport system the organic coolants can leave deposits on nearby surfaces, thus affecting heat transfer and flow characteristics. Following a large loss-of-coolant accident the containment can become filled with combustible vapors requiring a specialized fire suppression system.

The choice of UC fuel introduces a concern about fuel reactions with water or air in certain accident scenarios.

3.3.3.1.3 Other Impacts—The higher neutron absorption of an organic coolant, in comparison with heavy water, can be offset by the smaller inventory of structural material required in a lower pressure system. It is not clear whether or not a DUPIC fuel cycle (burning spent LWR fuel in a CANDU) could be achieved with a CANDU-OCR.

3.3.3.2 Generation IV Goals—Capabilities of OCRs.

3.3.3.2.1 Sustainability Aspects—*Sustainability-1*—CANDU-OCRs would be better than the reference due to the higher conversion fuels cycle options. There is little uncertainty due to the experience with CANDU-PHWRs.

Sustainability-2—CANDU-OCRs would be similar to the reference. There is some uncertainty—as better than reference might be achievable if DUPIC fuel cycles were viable with CANDU-OCRs.

Sustainability-3—CANDU-OCRs would be slightly worse than the reference due primarily to the theoretical opportunity to maliciously take advantage of the online refueling capability for the production of plutonium in low burn-up fuel bundles.

3.3.3.2.2 Safety and Reliability Aspects—*Safety & Reliability-1*—CANDU-OCRs would be much better than the reference because of the lower activation of primary coolant. Many areas of the primary coolant loop might be accessible for maintenance and inspection during operation. The working group average score was only "better" than the reference, perhaps indicating an uncertainty over safety that might be allayed if the detailed design was better understood.

Safety & Reliability-2—CANDU-OCRs would be similar to the reference since the small positive reactivity coefficients are offset by the lower voiding rate of the low-pressure organic coolant.

Safety & Reliability-3—CANDU-OCRs would be similar to the reference. The improved retention of fission products in the UC fuel is partially offset by the possibility of fuel reactions with air/water and coolant fires.

3.3.3.2.3 Economic Aspects—*Economics-1*—CANDU-OCRs would be better than the reference. The low production costs of current CANDU-PHWRs would be improved by various changes in the plant design including a single fueling machine, lower pressure system and lower activity coolant loop. The same comment applies here as for safety goal #1, that the working group average score rated OCRs "similar" to the reference even though the advocate assessment implied the economic metrics would be better than the reference LWRs.

Economics-2—CANDU-OCRs would be better than the reference due primarily to the lower cost of the coolant (compared to other advanced cooling concepts) and the savings due to the low-pressure coolant. Again, the same comment applies here as for safety goal #1, that the working group average score rated OCRs "similar" to the reference even though the advocate assessment implied the economic metrics would be better than the reference LWRs. The organic coolant might be perceived as more challenging to work with than conventional light water coolant, or more expensive to develop to meet Generation IV goal metrics.

3.3.3.3 Research and Development Challenges for OCR. The OCR concept report identified a number of research and development needs. Although the development of CANDU-OCR reactors was halted by AECL, their OCR test reactor, WR-1, continued operation for 10 years beyond the reference reports publication. It is possible that some of the issues have been studied further, resolved, or rendered irrelevant by this and coincidentally relevant developments in the CANDU-PHWR program. The major R&D needs for CANDU-OCR systems are as follows.

- UC reactions with air and water—increased testing to confirm and validate knowledge of fission product release rates during UC reactions with air/water.
- Fuel defect detection—the increased fission product retention makes detection of failed fuel more difficult. Detection of failed fuel is extremely important because such a plant would be operated under the assumption that the primary coolant loop had a low activity.
- Coolant flammability and fire suppression—investigation and testing of fire suppression systems to ensure feasibility at appropriate scales.
- Coolant fouling—better understanding of coking and fouling behavior of organic coolant is required at all possible flow velocities.
- Coolant blowdown—new experiments are needed to confirm the blowdown rates and their sensitivity to scale effects.
- Reactivity coefficients—further analysis of the reactivity coefficients is necessary to ensure that all control and safety systems will respond as necessary.

3.3.4 Concept Viability Evaluation for OCR

The score sheet developed through a detailed concept presentation and discussion within the Nonclassical Concept TWG is presented in Table 12. A previous TWG-4 meeting established the score sheet, based upon consensus. It is intended as a qualitative screening of the concepts potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

The decision of the TWG-4 at this time is to retain the OCR concept for further Generation IV evaluation.

Table 12. Qualitative Summary Evaluation of the OCRs.

Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	+	• Higher conversion ratio
Sustainability 2 Waste Management	=	• Similar to CANDU
Sustainability 3 Weapons Proliferation Resistance	=	• Online processing vulnerability
Safety and Reliability 1 Worker Safety and Plant Reliability	+	• Superior coolant properties • Low pressure operation • CANDU experience
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	=	• Lower void rate
Safety and Reliability 3 Mitigation against Offsite Emergency	=	• Retention of FPs in fuel
Economics 1 Life-Cycle Cost Advantage	=	• Single fueling • Lower activity coolant loop
Economics 2 Low Financial Risk	=	• Low cost coolant (although higher than LWR) • Lower pressure operation

An additional, more quantitative, summary score sheet of the scoring for potential to meet or exceed Generation IV goals is presented for the organic cooled reactor concept in Table 13. To compile the statistics for this table, the members of the Non-Classical technical working group were independently polled on all the categories defined by the Gen IV evaluation and methodology group. Under each goal there were a number of criteria and metrics, all of which were scored, but only the category averages are presented in Table 13. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean.

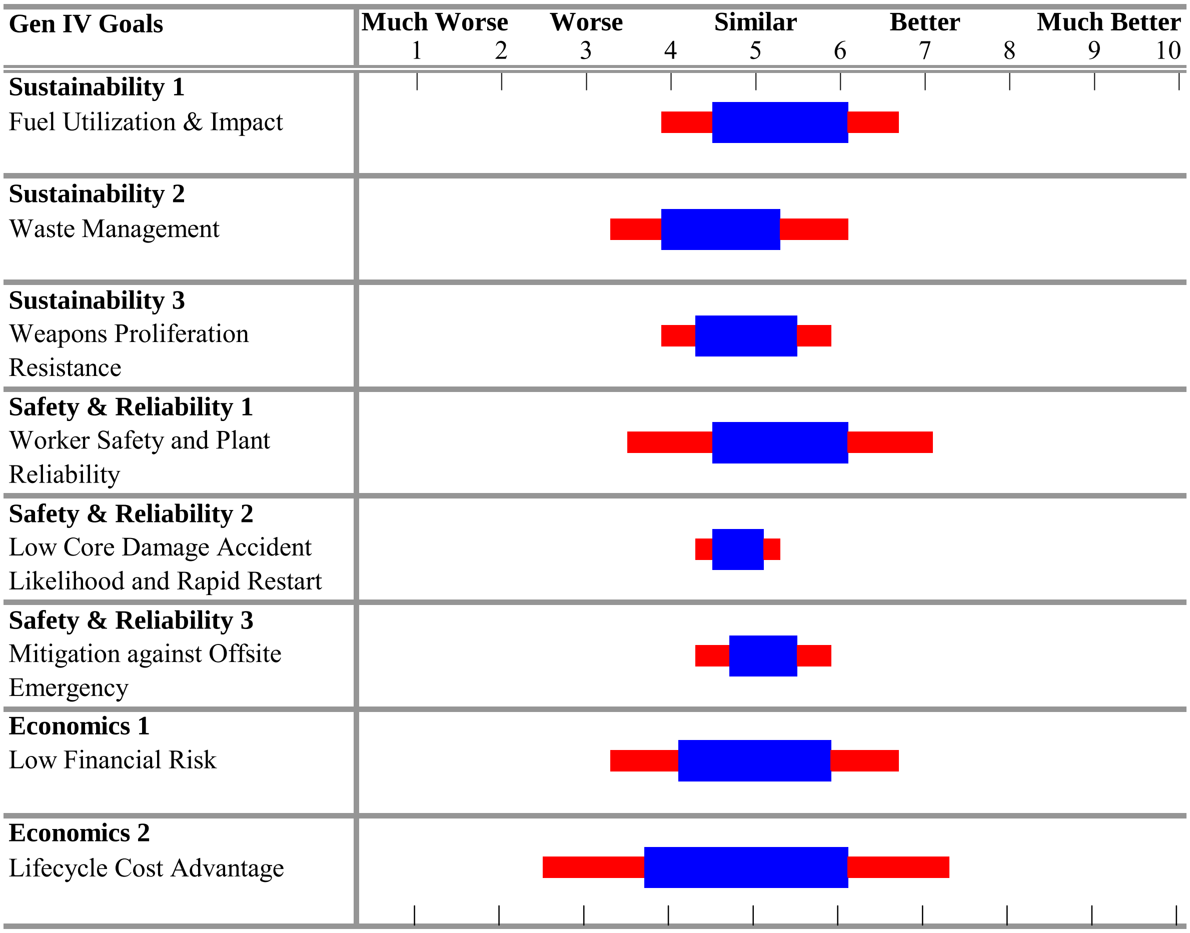
The large spread of scores indicated in this table reflects both the uncertainty in the data that was available for the 14 TWG-4 members to base their assessments upon, as well as the expanded scale that now ranges from 1–10, rather than the five-fold (– –,–,=,+,++) scale used in the qualitative consensus-based screening.

3.3.5 Metallic-Vapor Cooled Reactors (FSEGT)

These informal entries in response to the Generation IV request for information were received after the deadline, but are included here because it was felt that the Gen IV Liquid Metal Cooled concept technical working group might not have time to evaluate these extra systems.

At the Obninsk Institute of Physics and Power Engineering (IPPE) advanced liquid metal cooled reactors are being developed. The SoBAR concept is a sodium cooled pool type reactor. LeBAR is a lead cooled version. DOBARA is an array of remotely operated reactors. SEBAR is a sodium evaporating reactor design. Besides having inherent safety features, low pressure, remote automation and control, these reactors are also associated with a specially designed IPPE gas turbine for liquid metal reactors with a high availability factor and essentially constant plant efficiency over a wide range of power ratings.

Table 13. Quantitative score sheet evaluation of the OCR concept.



They form the basis of four projects currently underway at IPPE. Comparatives are drawn between these projects in Table 14. The projects are as follows: (1) Fast Sodium cooled two-circuit reactor with Gas Turbine (FSGT); (2) Fast Lead cooled two-circuit reactor with Gas Turbine (FLGT); these two concepts properly belong with the Liquid Metal-Cooled Reactors Technical Working Group report—TWG3—(which was understood to be already quite extensive), so they are included in this report because they failed to make the deadline for TWG-3, and are discussed in Section 5. The other two concepts are: (3) Distantly Operated Reactor Complex (DORC) which is discussed in Section 4.3.3 under the "Modular Deployable Reactors" heading, and (4) Fast Sodium Evaporating two-circuit reactor with Gas Turbine (FSEGT). This last concept fits into the present Nonconventional Coolant category as described next.

3.3.5.1 Fast Sodium Evaporating Two-Circuit Reactor with Gas Turbine (FSEGT). The FSEGT reactor concept (Figure 9) is based on the technology of sodium evaporation.[24] After 17 years of investigation, technical solutions for use of sodium evaporation from a two-phase mixture have been developed. The key point of this concept is the two-phase mixture coupled with sodium evaporation. Such a process is similar to that employed for liquid metal steam generators used for space nuclear power units in some former Russian projects.

Table 14. Comparison of IPPE liquid metal cooled concepts.

Project	Cladding Material	Cladding Temperature	Turbine's Gas Temperature	Turbine Efficiency
FSGT step 1	Existing stainless steel	750°C	650°C	45%
FSGT step 2	High temperature alloy to be tested	1,100–1,200°C	1,100–1,200°C	~60%
FLGT step 1	Existing stainless steel	650°C	550°C	42%
FLGT step 2 (without pumps)	Existing stainless steel	650°C	550°C	42.5%
FLGT step 3	High temperature ceramics	1,200–1,300 C	1,050–1,150 C	~60%
Sodium based DORC	Same as FSGT	Same as FSGT	Same as FSGT	≈ 2% lower than the FSGT
Pb-based DORC	Same as FLGT	Same as FLGT	Same as FLGT	≈ 2% lower than the FLGT
FSEGT	Existing stainless steel	800°C	750°C	47%

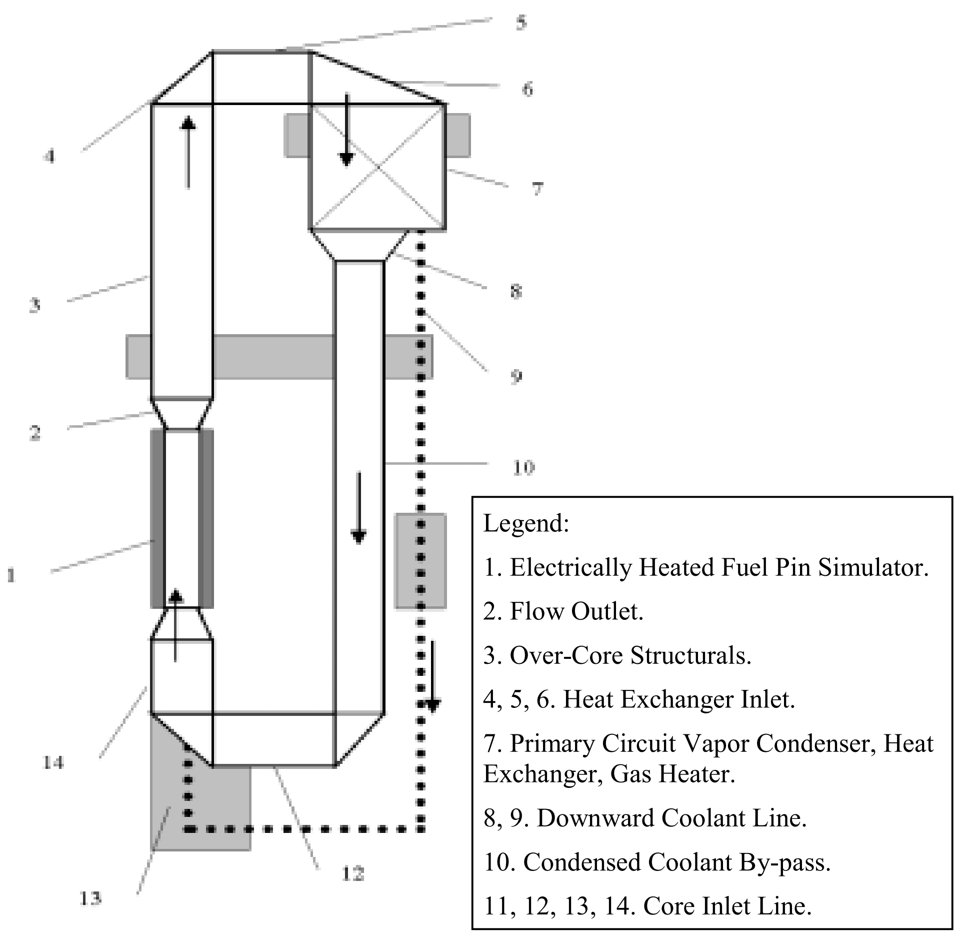


Figure 9. FSEGT experimental model layout. (Drawing courtesy of IPPE.)

3.3.5.1.1 Overview of the FSEGT Program—Since the mid-1960s there have been a number of attempts to apply liquid metal heatpipes to reactor technology. The benefits are well known: "pumpless" coolant circulation, relatively low pressure operation, small coolant inventory, and others. But by being subjected to heavy irradiation the materials change properties, including shape, and chemical elements content. Unfortunately, a necessary requirement of the heatpipe concept is strong shape stability with precision level about one-hundredth of a millimeter. Corrosion and chemical changes also represent serious heatpipe wick resource limitations.

A precisely manufactured wick is not the only way to force thin liquid layers to stay on the fuel pin surface. A wickless liquid metal evaporating system construction would have resource of the same level the conventional reactors has. Fast Sodium Evaporating two-circuit reactor with Gas Turbine—FSEGT concept is based on technology of wickless sodium evaporation. Experiments have led to some technical solutions for using sodium evaporation from a two-phase mixture. It is steadier than the direct sodium boiling method. Liquid sodium overheating and explosion-like sodium boiling modes are eliminated by a number of innovations, including cooling mixture preparation, coolant flow optimization, and others. On the other hand, such sodium evaporation equipment is simpler, more reliable, and easier to manufacture than heatpipe based evaporation proposals.

Sodium evaporates at significantly lower pressure than water, thus reducing the primary vessel requirements. At the same time, the sodium evaporation approach allows for higher operational temperatures and more stable exit temperatures than a system using sodium boiling. Use of sodium evaporation allows significant reactor liquid metal coolant inventory reduction. Like some boiling water reactors the FSEGT primary circuit can operate over a wide range of the design power with no external pumps. Attempts are being made to increase the coolant's natural circulation level up to 100% of reactor power or even more for emergencies.

The sodium evaporation approach (unlike conventional single phase sodium systems) requires significantly larger void space in the primary circuit to assure stable primary loop flow. This means that the evaporation process is very sensitive to the sodium inventory in the primary circuit. Any ingress of additional sodium into the primary loop through leaks from sodium storage and makeup tanks or from the intermediate loop can adversely affect the stability of the evaporation process. Eliminating the intermediate sodium loop and directly using gas in the secondary system, which converts the energy through a gas turbine, can avoid these potentially detrimental effects. This is the approach taken in the FSEGT design. By directly transferring the energy to a gas loop, this leads to higher gas temperatures and the system efficiency is about 2% higher than in the FSGT design (Section 5.1.2).

The Fast Sodium Evaporating reactor with Gas Turbine will have additional increase in efficiency, safety and reliability due to extensive use of passive principles in the coolant circulation system as well as simplification of equipment. Current work presumes usage of a temperature level of about 800°C and proven sodium-tolerant stainless steel. Due to the complete absence of wick effects, FSEGT material limitations are set only by the behavior of strength properties affected by irradiation. Therefore, there are a number of materials that may be suitable for leveraging further temperature increases. But for testing FSEGT principles the steel is found to be sufficient.

3.3.5.2 Generation IV Goals—Capabilities of the FSEGT Concept. Comments on Generation IV goals were not itemized in the advocate summary submission that the TWG-4 was given for this concept. However, the following comments on sustainability, safety, and economics issues could be gleaned from the discussion.

3.3.5.2.1 Sustainability Aspects—Sustainability Goals 1 and 2: The fuel cycle already supposes possible flexibility. If some country's local laws prohibit closed cycles, fuel cell design would allow long term storage for every program mentioned above. But ultimately, every fuel cycle has to be closed, when environmental impact is considered, whether the power plant itself operates open or closed.

3.3.5.2.2 Safety and Reliability Features—*Safety and Reliability 1*—Sodium evaporation cooling affords greater stability than direct sodium boiling. Liquid sodium overheating and unstable energetic sodium boiling modes are eliminated by a number of innovations including cooling mixture preparation, coolant flow optimization, and others. Natural circulation eliminates the need for circulation pumps. Sodium evaporation equipment is simpler, more reliable and easier to manufacture than heatpipe based evaporation proposals. Unlike water, sodium evaporates at significantly lower pressures, thus reducing the primary vessel structural requirements. At the same time sodium evaporation allows for higher operational temperatures and more stable exit temperatures. Use of sodium evaporation provides for a significant reduction in the liquid metal coolant inventory. Like some boiling water reactors the FSEGT primary circuit can operate over a wide range of the designed power spectrum with no external pumps. Attempts are being made to increase the coolant's natural circulation level up to 100% of the reactor's power or even more for emergencies. The aforementioned gas turbine energy extraction vacates the need for hazardous sodium loops.

Safety and Reliability Goals 2 and 3—The FSEGT reactors should have no worse core damage risk and offsite emergency response needs than existing conventional nuclear power plants.

3.3.5.2.3 Economic Aspects—*Economic Goals 1 and 2*—The FSEGT will have an additional increase in efficiency, safety and reliability due to extensive use of passive principles in the coolant circulation system and equipment simplification. In view of current international safety regulations, the trend in nuclear reactor design and construction has to be directed towards not only designing systems as safe as possible but also towards the most economically efficient design possible within the safety rules. The programs mentioned above could potentially satisfy the most stringent safety criteria. Most importantly the FSEGT concept could lead to improved system performance due to increased turbine efficiency. This would make the concept very competitive as a Generation IV design.

3.3.5.3 Research and Development Challenges—FSEGT. IPPE has a unique sodium evaporation test facility. Currently the work on sodium evaporation is approaching prototypic FSEGT thermo-hydraulic conditions that will demonstrate the engineering feasibility of the technology. This is needed to establish the technical feasibility of the concept. It remains to be clarified whether it will be sodium or some alloy of sodium that will be used as coolant, and what type of heat exchanger must be used. Two different types are presently under consideration.

3.3.6 Concept Viability Evaluation for FSEGT Reactors

Scoring for screening potential is omitted for this concept owing to its late inclusion in the request for information. Essentially the technical working group members have not had enough time to fairly and in an unbiased manner assess this concept.

3.4 Non-Convection Cooled Reactors

Number	Concept Name	Sponsorship
N15	Solid State Heatpipe Cooled Reactor	Oregon State University

The Non-Convection Cooled Reactor concept set includes all reactor designs that do not use bulk convective cooling to transport the core power. Reactors in this concept set are designed to either conductively or radiatively transfer heat either to a passive heat transport medium, such as a heatpipe, which transfers the heat to a power conversion system, such as a thermoelectric converter or Stirling engine, or directly to a static in-core energy conversion system such as a thermionic diode.

Reactors in this concept set are characterized by relatively high operation temperatures and have the potential to contribute to high overall power conversion efficiency if the use of the high temperature heat can be adequately used for an in-core power topping cycle, which would then be coupled to a low temperature conversion system. These concepts are promising in two ways: they have the potential for increasing the overall system efficiency, or they can be used as a special purpose reactor for remote application. It is unlikely that they will be deployable under the Generation IV guidelines for large-scale power applications. Because of this promise for special purpose application these concepts should continue to receive attention and be deployed if and when these applications are needed. One specific nonconvection cooled reactor design is discussed below.

3.4.1 Solid Core, Solid State, and Heatpipe Cooled Reactors

This set of reactors and systems[25] generally includes no direct working fluid to actively cool the reactor core. Reactors in this class include concepts that use only conduction to remove the fission-generated heat from the reactor core, systems that use heatpipes to remove the heat from the core, systems that conduct and directly radiate the heat from the reactor core, and systems that utilize in-core energy conversion to generate part or all of the electricity that they produce. These concepts have been typically explored in low capacity reactors for space applications and their power levels have been in the 1 Kw to 5 MW electrical power output range. With modification they could also be considered for remote terrestrial applications where small amounts of electricity would be needed. They would generally not be considered for large-scale electricity production because their maximum power output is severely limited by the short conduction and heat transport paths that are fundamental to their design. It could easily be expected that their deployment costs may be prohibitive for applications outside of specialized remote locations. The important cost consideration that needs to be determined is whether it is less expensive to deploy and operate such a remote operating reactor system than it is to deploy the necessary transmission lines to bring the power in from another location or deliver other more conventional power supplies such as diesel generators and their required fuel supplies. One can envision applications where many tens to hundreds of kilometers of power lines are needed to reach remote sites where it would be impossible to deliver power lines, or extremely costly to provide a regular supply of fossil or hydrogen fuels.

3.4.1.1 Generation IV Goals—Capabilities of Solid State Heatpipe-Cooled Concept. These are really concepts that fit a remote siting niche, in which case the only advantage that they could play would be in terms of cost. They may be useful in potential applications that need products other than electricity, such as hydrogen, high temperature process heat, desalinization, or in other applications that are not known today. They may also improve the proliferation resistance of a nuclear system if they could be designed to be entirely self contained so that they could be delivered to the remote site in a ready to operate configuration and then retrieved at the end of their useful lifetime. Otherwise, they are not generally recommended for large-scale power production.

3.4.1.2 Research and Development Challenges—Solid State Heatpipe. These concepts have received considerable development in the space nuclear power research community. Some of this technology could be straightforward to implement, but probably needs at least another 10 to 20 years of development before it could be deployed in a terrestrial application. Specific needs are in the area of high temperature fuels development, high temperature structural materials, specialized materials with appropriate work functions for thermionic emitters and collectors, high emissivity coatings, highly conductive materials, high temperature, long-life heatpipes, and long-term autonomous operational dynamics. Design issues arise in many systems when these concepts are considered for terrestrial application. In space applications they do not have containment, for example.

3.4.2 Concept Viability Evaluation for Non-Convection Cooled Reactors

The score sheet developed through a detailed concept presentation and discussion within TWG-4 is presented in Table 15. A previous TWG-4 meeting established the score sheet, based on consensus. It is intended as a qualitative screening of the concept's potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

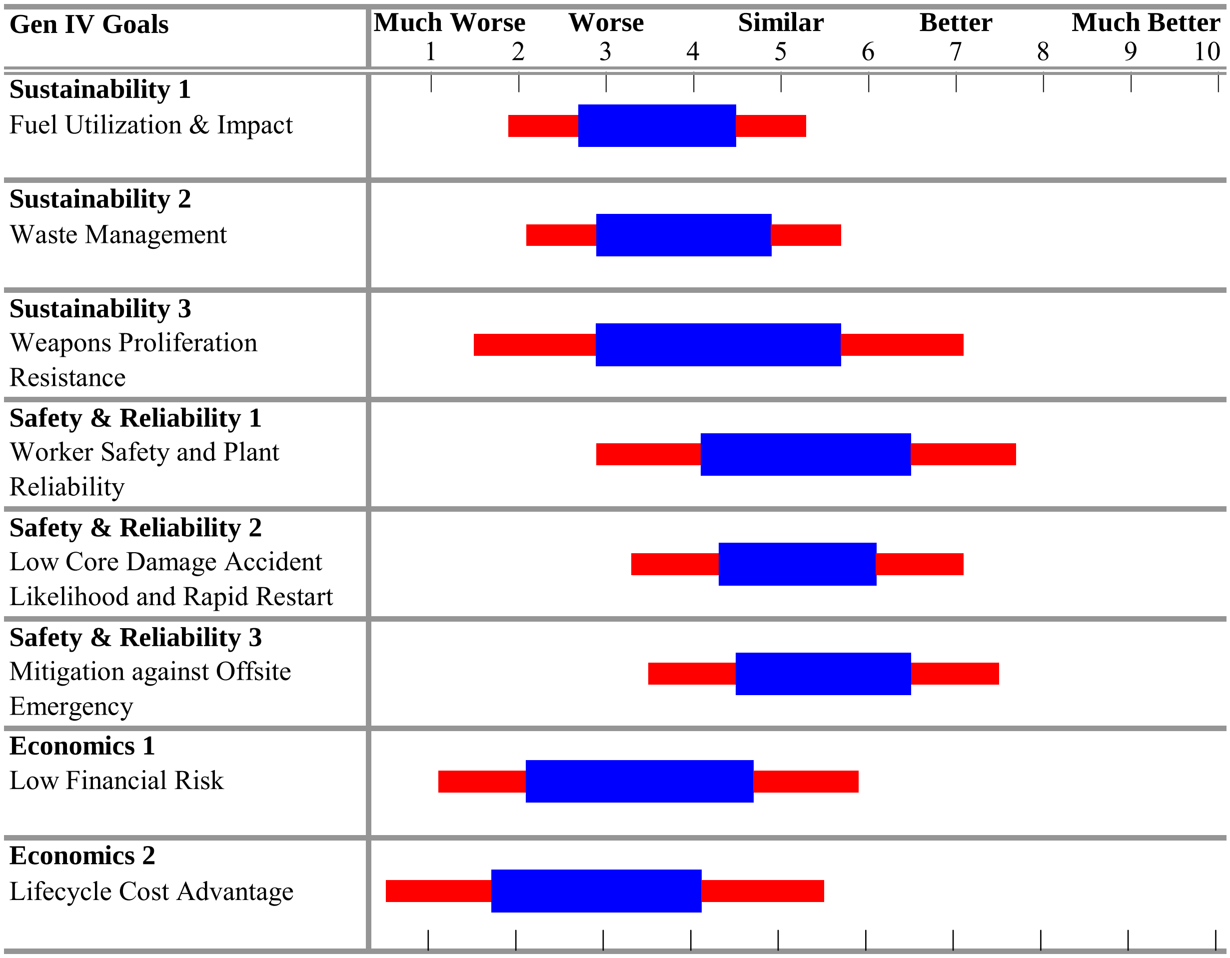
Table 15. Qualitative Summary Evaluation—Non-Convection Cooled Reactor Set.

Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	–	• Multiple fuel options – Power per kg of fissile fuel lower than LWRs
Sustainability 2 Waste Management	–	• Potentially low inventory – Fuel cycle less effective than LWRs
Sustainability 3 Weapons Proliferation Resistance	–	• Low inventory – Higher enrichment
Safety and Reliability 1 Worker Safety and Plant Reliability	+	• Passive heat removal • Low radiation exposure
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	+	• Stable energy conversion
Safety and Reliability 3 Mitigation against Offsite Emergency	+	• Small scale production only
Economics 1 Life-Cycle Cost Advantage	–	• Smaller secondary loop • Space power applications (niche markets, although costs may be higher in absolute sense) • Improved conversion efficiency • Lower activity coolant loop
Economics 2 Low Financial Risk	–	• Huge cost benefit for remote site applications • Capital cost advantages – Absolute costs probably greater than LWRs

The TWG-4 decision was that the solid-state heatpipe-cooled concept, or Non-Convection Cooled Reactor (NCCR), should be considered for specialized uses and has no significant potential for Generation IV large-scale power production. No R&D plan development will be considered for the concept at this stage.

An additional, more quantitative, summary score sheet of the scoring for potential to meet or exceed Generation IV goals is presented for the organic cooled reactor concept in Table 16. To compile the statistics here, the members of the TWG-4 were independently polled on all the categories defined by the Generation IV evaluation and methodology group. Under each goal there were a number of criteria and metrics, all of which were scored, but only the category averages are presented in Table 16. The

Table 16. Quantitative Score Sheet Evaluation of the NCCR Concept.



central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean.

The large spread of scores indicated in this table reflects both the uncertainty in the data that was available for the 14 TWG-4 members to base their assessments upon, as well as the expanded scale that now ranges from 1–10, rather than the five-fold (– –,–,=,+,++) scale used in the qualitative consensus-based screening.

3.5 Direct Energy Conversion Reactors

Number	Concept Name	Sponsorship
N9	QFMC —Quasi-Spherical Fission Magnetic Cell Array	SNL
N12	FFMC—Fission Fragment Magnetic Collimator	SNL

Direct Energy Conversion reactors[b] are built from very thin uranium dioxide foil fuel. Fission fragments carry most of their energy and electric charge out of the microns thick foils to direct power conversion devices. There are a multiplicity of magnetic field configurations and high-voltage electric fields that can be used to convert the kinetic energy of the charged fission fragments to electric current. The energy conversion ratio for these reactors is a strong function of the thinness of the fuel foil. This design constraint competes with criticality and burn-up requirements. Two engineering arrangements have been proposed: a spherical unit cell dispersed in a moderating material,[26] and a cooled filament array that uses a separate magnetic collector to capture the fission fragments.[27]

3.5.1.1 Feasibility Issues for QSMC and FFMC. Important technological and feasibility issues affecting both concepts (to be described below) are questions concerning:

- Achievable efficiency of fission fragment collection (cathode release fraction, electron capture by fission fragments).
- Integration of a magnetic field system into the cell/array.
- Cooling of the electrodes.
- Feasibility of a critical assembly (fissile material/moderator ratio, poisoning effect of structures).

Resolution of these challenges will have an impact on the feasible efficiency. Specific studies are required to investigate and evaluate these aspects. The concepts offer the potential for compact, high-energy conversion efficiency, power systems, but there are a number of significant engineering challenges. Development work is required before the concept can be compared with developed systems. The achievable efficiency may not be greater than direct cycle systems. The development timetable may not be compatible with the Generation IV objectives. Also, the highly enriched fuel is a concern from a proliferation risk perspective.

3.5.2 Quasi-Spherical Magnetic Cell (QSMC)

3.5.2.1 Description of the Concept. The Magnetically Insulated Quasi-Spherical Fission Cell[26] and the Fission Fragment Magnetic Collimator (see Section 3.5.3 below) are fission product direct energy conversion concepts aimed at collecting the fission energy directly from the fission fragments as an electrical current. The overall basic arrangement is to use a very thin cathode of fissionable material, a magnetic field to manage the polluting free electrons, a vacuum between the anode and cathode, and a high voltage anode to collect the positively charged fission products. The anode and cathode are connected to an external electrical circuit.

b. MHD generators can also be classed as direct energy conversion devices. These are discussed in Section 3.2.1 in the context of gaseous and vapor core reactors and in Sections 4.5.2 and 4.5.3 in the context of alternative power cycles.

In order to provide an effective power source, the basic arrangement (or cell) must be repeated in an array in order to form a critical assembly (see Figure 10). Since the amount of fissile material in each

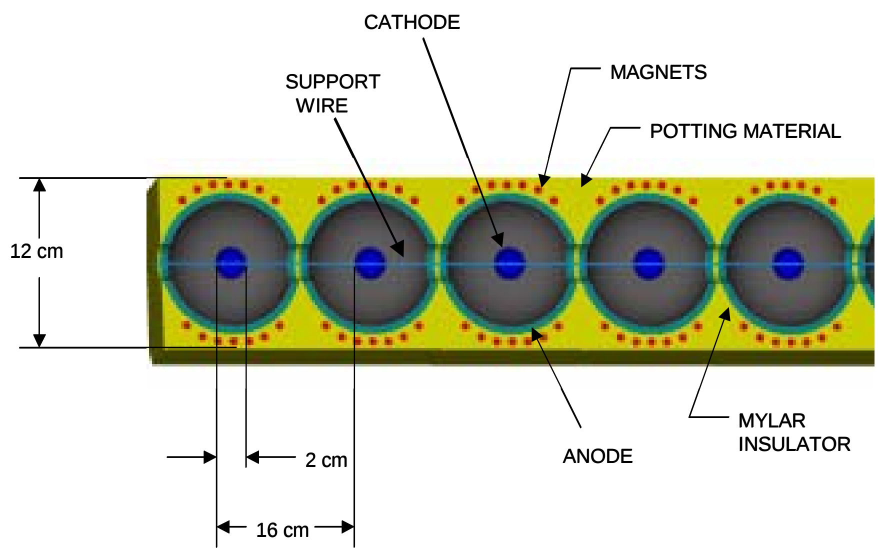


Figure 10. String of fission electric cells (each cell could produce 0.4 Watts of electrical energy). (Drawing courtesy of SNL.)

cathode must be small the fissile isotopic enrichment must be high (close to 100%) and a very effective moderating material must be used between the cells. The amount of fissile material required is very small when compared with conventional reactor systems. The actual amount required will depend on the detailed engineering but in particular will need to be sufficient to overcome the poisoning effect of structures, particularly the magnetic field system.

The field coils need to be constructed of a material that is superconducting at high magnetic field strengths, such as niobium. Such materials absorb neutrons and thus increase the mass of fissile material that will be required. Operation at one-half of the power density imposed by radiative cooling of the cathode would allow a 1 GW electric reactor with a fuel inventory of approximately 80 kg. The reactor volume would be 310 m^3 or a sphere of approximately 4.2 meter radius.

3.5.2.2 Generation IV Goals—Capabilities of Fission Electric Cells.

3.5.2.2.1 Sustainability Aspects—Refueling is envisaged by constructing the critical array in groups of cells plus moderator with each group being replaceable by relatively conventional means. Since very high enrichment fuel is envisaged the fuel cycle requirements are quite specific although the amounts are relatively small. In particular it will be necessary to have frequent refueling (several times a year) to compensate for burn-up.

3.5.2.2.2 Safety and Reliability Aspects— The safety assessment for this concept has not been detailed at this stage. However the basic safety concepts used in other reactor concepts are likely to be adaptable to this direct energy conversion concept. In particular moderator poisoning or specific absorber rods in the moderator can be envisaged as methods of control. The system is barely critical and so it is difficult to envisage significant reactivity faults.

3.5.2.2.3 Economic Aspects—This concept does not require the conventional engineering plant associated with a steam cycle. However the magnetic field system is likely to add significant cost.

Also heat deposition in the magnetic field system will be significant with a consequent need for a specific heat removal system. This will have an adverse effect on the energy balance of the concept. A small fraction of the released fission fragments impinging normally to the electric field with just the right velocity will have 100% conversion efficiency; anode heating from the leftover kinetic energy of some fragments may be recovered by cooling heat exchange. The system is robust and has no moving parts.

3.5.2.3 Research and Development Challenges—QSMC. Very low areal density materials need to be fabricated (<0.004 kgm^{-2} of fissile fuel coating). The cathode geometry is delicate and requires advanced materials engineering, two available methods are (1) edge supported thin films made by sputtering or vapor deposition onto dissolvable plastic film with bonding to a stainless steel support ring, or, (2) self-supporting foam shells manufactured from plastic and then burnt into carbon. Experience for such fabrication techniques are already available from other R&D programs. A large system bulk must be engineered owing to the trade-off with low fuel density and criticality. Even if criticality can be established within reasonable engineering parameters, it remains to be seen whether this device could produce enough electricity at cost to become commercially viable. However, overall, this concept is expected to be more suited to a specialist application rather than as a large power source. It is therefore proposed that it is considered in another context, other than Generation IV.

3.5.3 Fission Fragment Magnetic Collimator (FFMC)

3.5.3.1 Description of the Concept. The same principle used by the QSMC is exploited, namely, conversion of charged fission fragment kinetic energy into electricity via cathode-to-anode collection and drawing off the resulting voltage across the load. The idea behind the fission fragment magnetic collimator device is to use a magnetic field to collimate the heavy positive ion fission fragments without loss of energy.[27] This is achieved by setting up a cylindrical array of fuel cell tubes gathered together into a sparsely separated hexagonal array. A solenoidal magnetic field is then applied so that the Lorentz force on the ion fragments will send them spiraling in helices out either end of the cylindrical array.

An efficient charge collection mechanism is the Venetian blind collector, illustrated in Figure 11. The electrons are repelled by a negative grid and drawn back to the cathode, while the positive fission fragments have a fairly high probability of escaping through the gaps in the blind's grid. The magnetic field diverges at the ends of the cylinder and hence the ion trajectories fan out like an exhaust plume, so the collector should be part of a spherical hemisphere, and for highest efficiency the ions should impinge upon the anode as close as possible to normal incidence. The fuel cell tube walls must be extremely thin to allow the fission fragments to escape from the fuel layer and journey to the anode; there must be enough fuel in total to achieve criticality; heat removal from the cathodes must be achieved without significant neutron absorption.

Two alternative FFMC reactor configurations have been suggested to deal with the isotropic emission of fission fragments and the unwanted electrons.

Water-Cooled Tube Configuration—In this design a solenoidal extraction helps align the paths of the fission fragments without energy loss. Figure 12 shows a possible configuration. The tubes are configured in an array that is aligned with a strong magnetic field.

Thin-Walled Tube Configuration—In the water-cooled tube configuration, half of the power is lost immediately due to fission fragments emitted into the tube rather than away from it. At the very most this lost energy can be partially recovered by a conventional thermodynamic heat cycle. This loss can be avoided if the tubes consist of a very thin foil of uranium metal (or perhaps oxide). Then the fission fragment heading into the tube could pass through it with only a moderate loss. This configuration also decreases the amount of loss from fragments emitted away from the tube since those destined to hit the tube later might survive passage through it. The foil tubes might be cooled with helium if the flow rate is not too large. Conditions can be found for the tube diameter and tube spacing that maximizes energy extraction and number of tubes in the reactor (the latter is desirable for criticality).

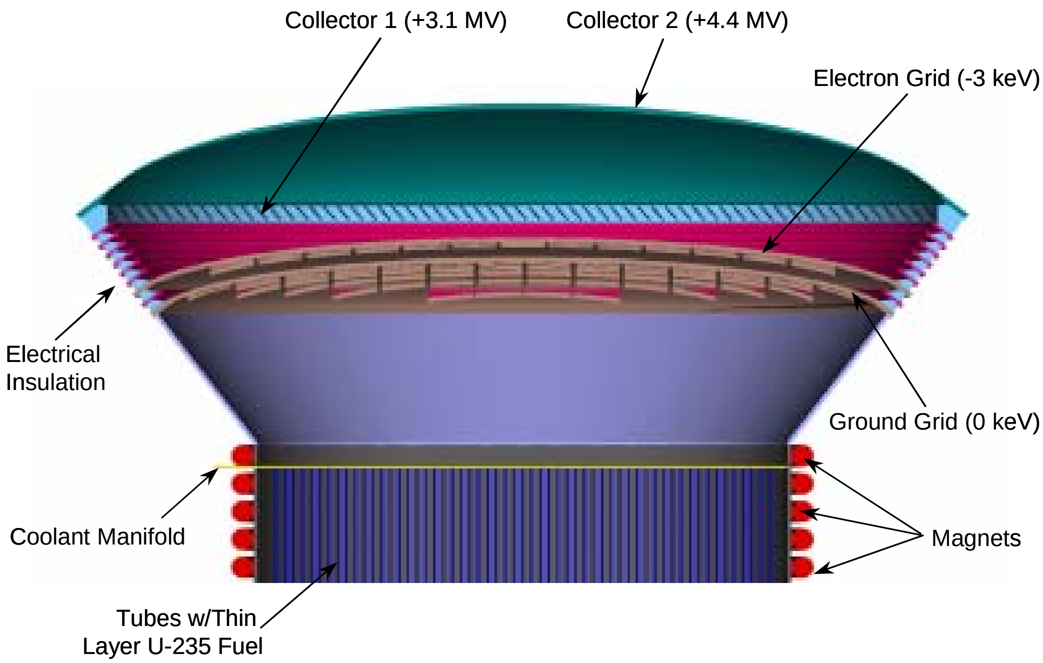


Figure 11. Two-stage Venetian-blind fission- electrical converter. (Drawing courtesy of SNL.)

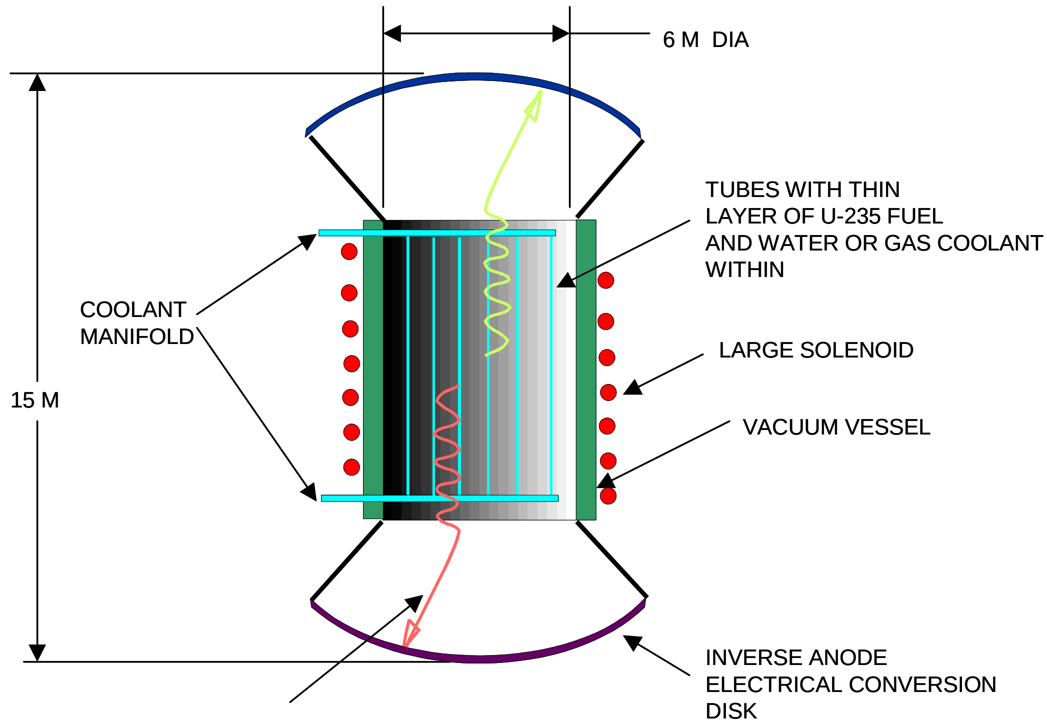


Figure 12. Possible configuration for a solenoidal DEC reactor. (Drawing courtesy of SNL.)

3.5.3.2 Generation IV Goals—Capabilities of the FFMC. The same arguments apply here as for the Quasi-Spherical Magnetically Insulated Cell, (see Section 3.5.2).

3.5.3.3 Research and Development Challenges—FFMC. Thin hollow-walled tubes can in theory be fabricated for reasonable fragment escape probability, and thus efficiency. An improvement in available magnet strength could be anticipated and deemed necessary. Engineering design is severely challenged to find a physically possible combination of thin fissioning fuel coating and high enough total fuel density to obtain reactor criticality and >30% electrical energy conversion efficiency. A large ($500\ m^3$) reactor with very thin walled tubes (≤1 μm) in preliminary analyses is the best design. The possibility exists for iterative design improvements to meet the performance goals required for commercial operation. The same comments apply to the Quasi-Spherical Magnetic Cell design. This concept should ideally be considered in another context since it is considered not to be feasible as a large power source.

3.5.4 Concept Viability Evaluation for Direct Energy Conversion Reactors

The score sheet developed through a detailed concept presentation and discussion within the Nonclassical Concept TWG is presented in Table 17. A previous TWG-4 meeting established the score sheet, based upon consensus. It is intended as a qualitative screening of the concept's potential to meet or exceed Generation IV goals. The table includes some brief comments that help to explain the rationale for the scoring. A score of (– –) is much worse than the reference, (–) is worse than the reference, (=) is similar to the reference, (+) is better than the reference, and (++) is much better than the reference light water reactor systems.

Table 17. Qualitative Group Summary Evaluation—DEC Set.

Generation IV Goals	Score	Qualitative Assessment

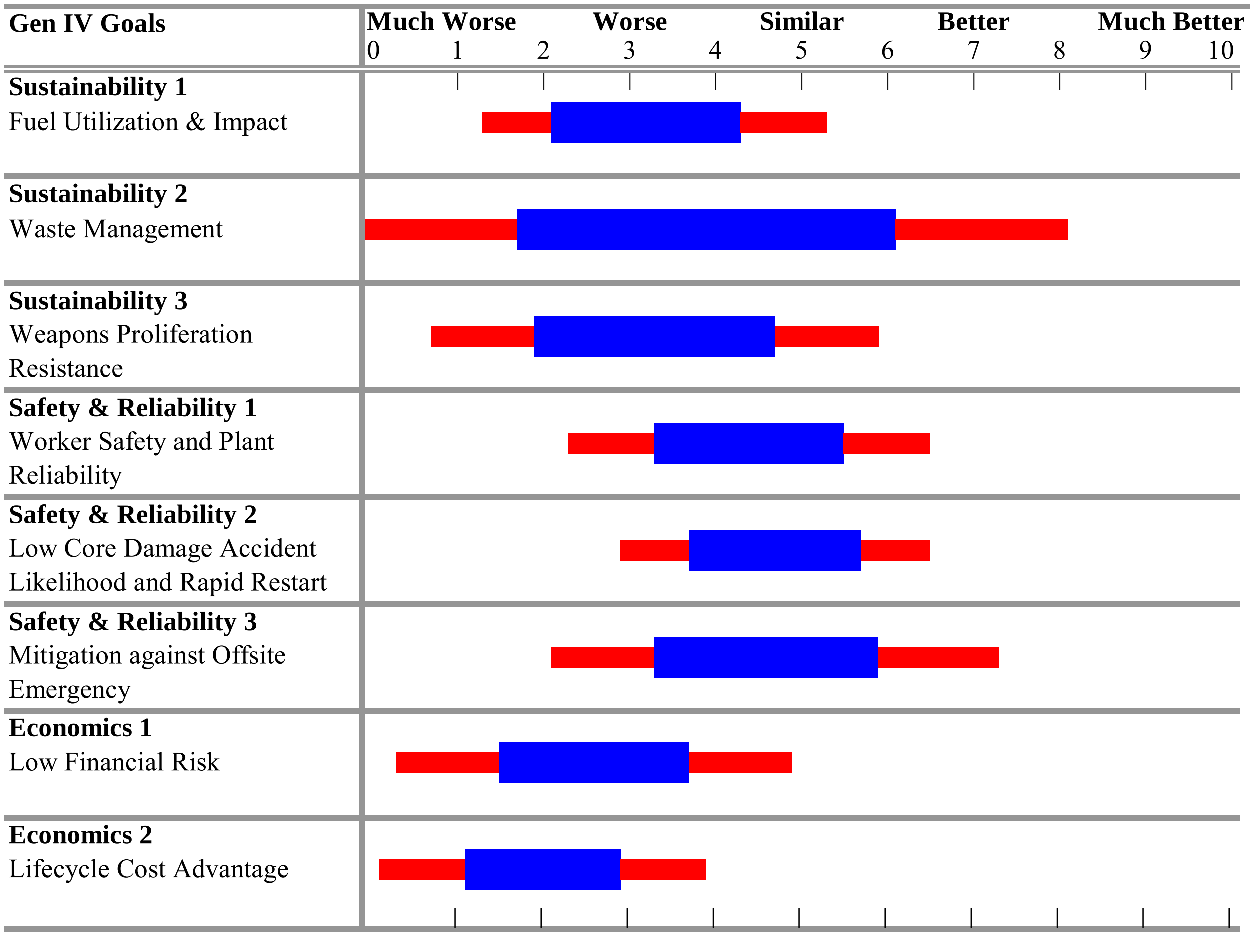
Generation IV Goals	Score	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	–	• Sparse fissile fuel – Only a fraction of the fission fragment's energy is used
Sustainability 2 Waste Management	–	• Lower effective fuel use compared to LWRs
Sustainability 3 Weapons Proliferation Resistance	–	• Difficult to extract Pu – Weapons grade fissile isotope is required
Safety and Reliability 1 Worker Safety and Plant Reliability	=	• No moving parts • Complete elimination of hydraulics
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	=	• Highly robust • Barely critical
Safety and Reliability 3 Mitigation against Offsite Emergency	=	• Barely critical
Economics 1 Life-Cycle Cost Advantage	– –	• Great scope for future design improvements – Difficult to recover R&D and initial capital cost
Economics 2 Low Financial Risk	– –	• Simple design, no moving parts – High financial risk

Although interesting on many counts, the TWG-4 decision was that these technologies are better considered for *specialized uses* and have no significant potential for Generation IV large scale power production. No R&D plan development will be considered for these concepts at this stage.

An additional, more quantitative, summary score sheet of the scoring for potential to meet or exceed Generation IV goals is presented for the organic cooled reactor concept in Table 17. To compile these statistics the members of the TWG-4 were independently polled on all the categories defined by the Generation IV evaluation and methodology group. Under each goal there were a number of criteria and metrics, all of which were scored, but only the category averages are presented in the Table 18. The central blue bars are spreads of one standard deviation about the mean. The red bars are spreads of 1.96 times the standard deviation about the mean.

The large spread of scores indicated in this table reflect both the uncertainty in the data that was available for the 14 TWG-4 members to base their assessments upon, as well as the expanded scale that now ranges from 1–10, rather than the five-fold (– –,–,=,+,++) scale used in the qualitative consensus-based screening.

Table 18. Quantitative Score Sheet Evaluation of the DEC Concept Group.


Gen IV Goals
Much Worse
Worse
Similar
Better
Much Better
0
1
2
3
4
5
6
7
8
9
10
Sustainability 1
Fuel Utilization & Impact
Sustainability 2
Waste Management
Sustainability 3
Weapons Proliferation Resistance
Safety & Reliability 1
Worker Safety and Plant Reliability
Safety & Reliability 2
Low Core Damage Accident Likelihood and Rapid Restart
Safety & Reliability 3
Mitigation against Offsite Emergency
Economics 1
Low Financial Risk
Economics 2
Lifecycle Cost Advantage

4. CROSSCUT DESIGN ISSUES

Nuclear power plant requirements have changed. There is a greater emphasis on safety, waste management, and public acceptance. Serious consideration is being given to destroying actinides rather than disposal in a repository. In a world of 10 billion people, much of the world will need large reactors with minimal resource requirements.

The technical breadth of the Nonclassical Concepts presented to TWG-4 has been substantial. The Gen IV Roadmap Crosscut Groups will evaluate additional concepts with specific crosscut design features. The list of crosscut design issues includes:

1. Minimum Waste Fuel Cycle

2. Diverse Energy Product Systems

3. Advanced Fuel Materials

4. Alternative Power Conversion Cycles.

These are discussed in turn below.

4.1 Minimum Waste Fuel Cycles

Number	Concept Name	Sponsorship
N1	CANDLE —Constant Axial Burn-up	Tokyo Institute of Technology
N17	MCPP—Multi-Component Power Plants	Kurchatov Institute, Russia

The four prerequisites for the continued use of nuclear power are generally regarded as safety, economics, nonproliferation, and waste disposal. Current and new reactor designs suggest that the first three criteria can and will be met. In addition, studies by the Fuel Cycle Crosscut Group indicate that the future use of nuclear power will not be resource limited, especially if the thorium-^{233}U cycle and the use of plutonium are considered. However, these same studies suggest that if and when acceptable permanent repositories can be sited and licensed, the number of these repositories needed to meet a growth in demand for nuclear power may be so large that they will become the limiting factor in the use of nuclear power. As the pressure on the back-end of the fuel cycle grows, there will be very significant pressures to reduce the volume of waste and the length of time that a repository must be monitored and meet performance criteria.

Minimum waste fuel cycles, or holistic fuel cycles,[28] have been receiving considerable attention during the last few years. This approach is characterized by considering, as a complete entity, the entire fuel cycle from mining and fabrication through reactor systems and reprocessing facilities to the waste stores. Historically all these facilities have been considered as separate entities and optimized to meet the specific function of each component. However, as noted above there are clear benefits and economic incentives to minimize waste, better use materials, and reduce dose. This will be achieved by considering the facilities as a complete system. Typically the issues to be considered are: (1) what is the best mix of reactor systems, (2) what is the best mix of fuel fabrication/processing systems, and (3) should the components still be considered separately or should a completely integrated plant system be considered? Scenario studies are required to investigate the options and the sensitivities of each aspect.

To reduce the amount of waste generated in the fuel cycle, several strategies can be implemented. Even at the front end of the fuel cycle, significant amounts of waste are generated in the mining and separation operations. Reduction of the volume of these materials will be achieved through fuel cycles that reduce the amount of feed material required through the uranium-plutonium system and the thorium-^{233}U cycles. The use of plutonium as an energy source will also be instrumental in utilizing the depleted uranium that now accumulates from the enrichment processes.

Very significant reductions in the volume and toxicity of the waste that must be stored in a repository are possible using advanced reactor and fuel cycle technologies. Several reactor concepts presented in this report, including those discussed in the following sections, will recycle the long-lived actinides in situ and convert them into fission products. A parallel strategy receiving much discussion but not covered in this report is the use of advanced accelerator applications that also transmute the transuranics into fission products with much shorter half-lives. The long-lived fission products such as cesium, iodine and strontium can be sequestered from the balance of the volume of waste and similarly treated. Finally and ultimately, other uses and processes might also be identified for components of the waste. Some of the same isotopes that are currently being produced by specialized reactors and accelerators for medical, industrial, and other applications are components of spent fuel. One potential example for significant quantities of these isotopes will be the need for food irradiation. Currently more than 50% of the food produced globally does not reach the consumer due to spoilage, and the loss of seafood is even higher. Electron beam irradiation is suitable for some applications, but for other needs the deeper penetration gammas from radioisotopes may be required. With the global population growing from the current 6 billion to 9 or 10 billion by midcentury and more and more land coming out of agriculture production worldwide, better use of the food resource will be critical to preventing massive shortages and starvation. Isotopes from spent fuel may play a key role in this, and materials currently considered to be a waste may in fact become a resource. When this develops, if it does, the pressure on repositories may be considerably eased and true minimum waste fuel cycles will be a reality..

4.1.1 Specific Generation IV Minimum Waste Concepts

This section introduces two particular concepts for efficient fuel use and waste minimization in some detail—the CANDLE concept and the Multi-Component MSR Power Structure. Space permits only a brief mention of how some of the other Generation IV concepts propose to minimize waste and maximize fuel use. Additional information for other Generation IV reactor concepts waste handling capabilities are given under each individual concept category above in Section 3.

4.1.1.1 CANDLE—Constant Axial Neutron Flux and Nuclide Density Burn-up. This reactor concept is designated CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy production). The intent of the CANDLE concept is to achieve a very long life, high burn-up system by a unique extended core design. CANDLE produces high burn-up due to a fast reactor spectrum with good neutron economy.[29,30] For the CANDLE burn-up strategy, a nuclear ignition region is set at the bottom or top of core (bottom for the case of Figure 13). Only natural uranium or possibly depleted uranium is charged in the remaining region. The distribution of each fuel nuclide in the ignition region can be obtained by numerical methods, and it satisfies the requirements for the ignition region. Therefore, an example nuclide distribution for ignition region can be gained by modifying the equilibrium distribution such as replacing short half-life nuclide by neutronically equivalent stable nuclide.

In the equilibrium state of CANDLE burn-up, distributions of nuclide densities, neutron flux, and power density move axially with corresponding constant shapes and the same constant speed along burn-up for constant power operation as shown in Figure 13. In the front region just before the burning region, natural uranium absorbs neutrons and ^{238}U is transformed to ^{239}Pu. In the burning region ^{239}Pu is

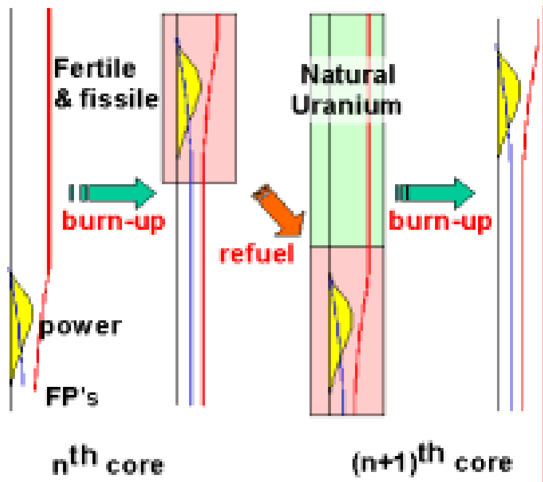


Figure 13. CANDLE Reactor Burn-up Strategy. (Drawing courtesy of Tokyo Inst. Tech.)

burning and produces neutrons and energy. The higher actinides are also produced in this region, and then not only ^{239}Pu but also other generated fissile nuclides will fission. The contribution of fast neutron induced fission is also considerable since the reactor is a fast reactor. From the front side to back side of the burning region the density of ^{238}U gradually decreases and the fission products (FPs) accumulate. In the region behind the burning region, the total density of FPs is so high to reduce local reactivity and to make power level negligibly low. In the front side of burning region ^{239}Pu is produced, and on the rear side of burning region FPs accumulate. These nuclide density changes make the burning region shift to the frontal direction.

4.1.1.2 Generation IV Goals—Capabilities of CANDLE.

4.1.1.2.1 Sustainability Aspects—The CANDLE concept is designed to achieve a very long life, high burn-up system. Preliminary calculations suggest that as high as 40% of the fertile material can be burned in a single core cycle. This projected long life core minimizes recycling and enrichment requirements, improving proliferation resistance of this system. The reactor vessel can be sealed for the life of the reactor. Preliminary calculations[31] indicate that the average burn-up of the spent fuel is about 40%, which means that 40% of the natural uranium is completely utilized without enrichment or recycling. The absence of high enrichment needs, other than for the one startup core, provides strong proliferation resistance.

4.1.1.2.2 Safety and Reliability Aspects—The safety and reliability of CANDLE[32] is better than the reference LWRs in that the system does not require any control system for compensating burn-up excess reactivity, the reactor is barely critical, and redistribution of fuel usually reduces its criticality performance, therefore, it is free from CDA accident. Reliability features in particular, as well as safety characteristics, are difficult to rigorously assess for CANDLE at present because the cooling system and heat generation characteristics are not well understood.

4.1.1.2.3 Economic Aspects—The CANDLE concept does not require large control margins to accommodate large reactivity swings during operation. It also does not require active flow distribution control during operation since the total power in each channel does not change over the life of the core. These features may allow simplification of these systems resulting in cost savings.

4.1.1.3 Research and Development Challenges—CANDLE. Although CANDLE appears to be an elegant concept, it will be a major challenge to demonstrate this technology. Potential pitfalls include achieving a healthy neutron economy without adversely affecting the core cooling performance or vice versa, and the difficulties of managing the material and structural implications of high burn-up percentages.

4.1.1.4 The MSR Multi-Component Power Architecture.

This concept originates from the Kurchatov Institute.[33,34] Its rationale comes from the realization that optimum fuel utilization for an integrated, mixed set of different reactor types, operating in a combined-system power plant might be better than any one single reactor type operating alone. The choice of nuclear power architecture for closing of the cycle and waste minimization was based on computational analysis of isotope flows for the following different variants of nuclear power systems: thermal light-water reactors, fast reactors, and molten salt reactors. A table of results, Table 19, suggests a system with thermal and molten salt reactors can be closed with minimum quantities of transuranic elements (TRU), but without fast breeders such a system either requires feeding by ^{235}U or should be based on a Th-U cycle. One option not yet studied is a closed MSR fuel cycle based upon a Th-U cycle.

If, however, fuel breeding in fast reactors remains a necessary condition for large-scale nuclear power development, then a three-component architecture can be considered, as in Figure 14.

This will burn minor actinides, Pu, Th and the long-lived fission products ^{99}Tc and ^{129}I in a molten salt reactor, with 40% total power in the U-Pu cycle produced by fast breeders (eventually eliminating the need for ^{235}U feed and thus minimizing economic and environmental impacts of proliferation and of uranium mining). About 50% of the total power is produced by thermal reactors (to minimize the plutonium used in the U-Pu cycle). Approximately 10% of the total power comes from critical or subcritical (accelerator driven system–ADS) molten salt burners (MSR) designed especially for closure of the fuel cycle on minor actinides.

Table 19. Quantities of TRU in Equilibrium Closed Fuel Cycles.

Reactor Types	LWR	75%LWR + 25% MSR	Fast (FR)	89 FR + 11% MSR	51% LWR + 38% FR + 11% MSR
Ave. Flux (# $cm^{-2}s^{-1}$)	10^{14}	10^{14} (LWR) 5×10^{15} (MSR)	10^{15}	10^{15} (FR) 5×10^{15} (MSR)	10^{14} (LWR), 10^{15} (FR) 5×10^{15} (MSR)
^{237}Np	0.72	0.12	0.11	0.02	0.02
Pu (Total)	5.8	3.1	21.1	18.0	10.4
$^{241}Am + ^{243}Am$	0.76	0.08	0.77	0.10	0.19
Cm (Total)	1.38	0.09	0.19	0.04	0.11
TRU (TOTAL)	8.7	3.4	22.2	18.2	10.7
TRU without Pu	2.88	0.30	1.10	0.17	0.3
Heavy Nuclide	265	283	121	117	221

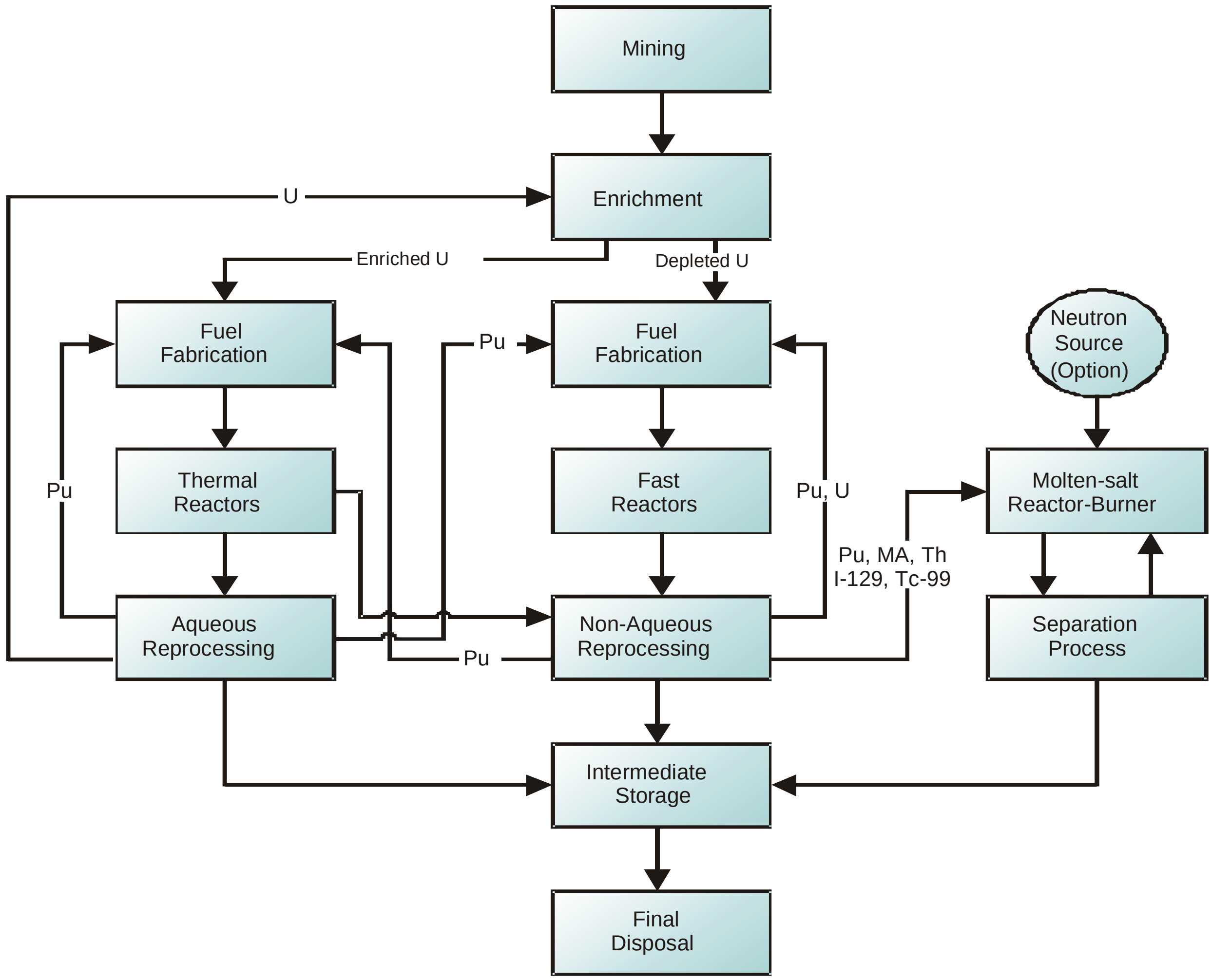


Figure 14. Kurchatov Institute Nuclear Power Future concept. (Drawing courtesy of ORNL.)

Another multicomponent system option is a subcritical accelerator driven MSR. The advantages of going to an MSR-ADS include exclusion of reactivity-initiated accidents. Reactor control becomes more effective, and additional neutrons are produced that allow a sufficient number of fissions of Np, Am, and Cm without adding ^{235}U, ^{233}U or Pu to the core, to decrease the required fraction of reactor-burners in the nuclear power section. But an ADS system also has disadvantages such as decrease in the nuclear energy fraction delivered to consumers, a complex design, lack of neutron sources meeting goals of reliability (7,000 hours reliability *per year for decades*), and possible deterioration of inherent safety with loss of primary coolant flow. The use of the cascade principle of neutron multiplication from the external source allows a decrease in input power, but increases the spatial power nonuniformity in the reactor.

4.1.1.5 *Generation IV Goals—Capabilities of Multi-Component MSR Concept.*

4.1.1.5.1 Sustainability Aspects—The MSR component used as a burner utilizes minor actinides and weapon-grade Pu, providing effective fuel utilization and sustainable energy generation. This also helps minimize the proliferation risk. When used as a power reactor the MSR with Th-U fuel cycle can provide neutron and nuclide balance necessary for minimization of uranium and thorium mining and ^{235}U utilization. Tuning the fraction of the total power delivered by each reactor in the multireactor system means the system as a whole can be optimized to minimize the toxicity and volume of long-lived radioactive wastes.

4.1.1.5.2 Safety and Reliability Aspects—The MSR has the natural safety advantage of fast negative density reactivity coefficient. Also passive drainage of molten fuel composition in subcritical tanks when temperature limits are exceeded in the primary circuit provides excellent handling of positive reactivity surges. Leakage accidents do not precipitate offsite emergency responses because releases are limited by freezing of the molten salt and because volatile radioactive waste gases in the fuel are removed online in reactor operation. However, consequences of reactor core damage still need to be analyzed.

4.1.1.5.3 Economic Aspects—Development of new fuel types and complexity of design are avoided. The structural overhead required to remove heat from the core is not required. The MSR system does not require new fuel materials or complex fuel element configurations. The heat removal system can potentially be simplified in comparison with other core designs. Also important to economy is the fact that average neutron flux density can be provided higher than 2×10^{15} neutrons cm^{-2} s^{-1}. Online removal of fuel for regeneration and removal of fission products and online feeding with actinides also contribute to MSR economy. The spent fuel problem is solved in MSR operation by constant online reactor operation, so there is no unpredictable postponement after reactor decommissioning.

4.1.1.6 Research and Development Challenges—MSR Multi-Component Power Structure. Problems that remain to be solved before such systems can come online include chemistry, fuel composition and separation, design of materials for molten salt containment, and resolution of physics modeling issues such as temperature distribution and density fluctuation in the salt, delayed neutron fractions, isotope kinetics, fission gas product removal, temporal-spatial distribution of radioactive and heat generation sources. A fully 3-D thermal hydraulics heat and mass flow code will need to be developed with volume power generation accounted for in all primary volumes.

4.1.2 Concept Viability Evaluation for Minimization of Waste Fuel Cycles

As the two specific concepts described above are actual reactor concepts they can be evaluated if additional information regarding their design characteristics becomes available.

These different uses of nuclear energy impose different requirements on nuclear energy facilities. Power plants producing electricity can also produce heat for other applications. Reactors used for just district heating may have unique safety requirements because of their very close proximity to large populations. Thermochemical hydrogen production, based on our current understanding, will require very high temperature heat.

From a broad perspective, nuclear energy can be used for three major applications that correspond to three major categories of energy use.

- *Electricity*—The primary historical use of nuclear power has been for the production of electricity. Almost all research, development, and planning has been directed at this application.

- *Heat*—Nuclear power produces heat that can be used for district heating and water desalting. Several nuclear power plants have been used to produce electricity and hot water for district heat. Heat from nuclear power plants has been proposed for desalting water.

- *Hydrogen*—Water can be converted to hydrogen and oxygen using thermochemical methods and very high-temperature heat. The world currently consumes ~50 million tons of hydrogen per year (200 GWth equivalent) for production of fertilizer, production of chemicals, and refining of oil to produce transport fuels. Hydrogen may ultimately be used directly as a transport fuel. Hydrogen is a means for using nuclear energy to meet transportation requirements where electricity can't be used. Because transportation is such a large component of total energy use, it may ultimately become a major use of nuclear energy.

The qualitative assessment in Table 20 is given for minimum waste reactor concepts as a whole group. Lack of specific reactor details make it difficult to provide individual complete score sheets for these concepts at this time.

Table 20. Qualitative Group Summary Evaluation—Minimum Waste Concept Set.

Generation IV Goals	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	• High burn-up • Low inventory • Thorium fueled breeder
Sustainability 2 Waste Management	• In situ waste removal • Low waste efflux
Sustainability 3 Weapons Proliferation Resistance	• Actinide burner • Natural uranium & thorium • High burn-up
Safety and Reliability 1 Worker Safety and Plant Reliability	• Natural circulation • In situ waste removal (nuclear dialysis) • Low pressure
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	• No fuel damage, only leakage worry • Core plug drainage
Safety and Reliability 3 Mitigation against Offsite Emergency	• Fuel-coolant immiscible • In situ removal of volatiles
Economics 1 Life-Cycle Cost Advantage	• Good conversion efficiency • Modular deployable
Economics 2 Low Financial Risk	• Current technology base • Simple designs • Some materials issues

4.2 Diverse Energy Product Reactors

Number	Concept Name	Sponsorship
–	Hydrogen Production Reactors	ORNL

The potential future use of nuclear energy to produce hydrogen requires a separate discussion because it may be the link that allows nuclear power to meet transportation requirements—a large and growing fraction of the total energy demand. Liquid fuels used for transportation account for 24% of the total world energy demand and 25% of the energy demand in the United States. In some high-technology states such as California, one-half of the total energy consumed is in the form of liquid fuels for transportation. The hydrogen market is also a nontrivial existing and near-term market. It is estimated that if the hydrogen needs of the United States by 2010 were to be met by using nuclear power (50% efficiency), the necessary energy input would exceed the energy production of all nuclear reactors in the United States today.

While the primary use of hydrogen today is for fertilizer production (ammonia), the most rapidly growing use is for the refining of oil into transport fuels (gasoline, diesel, jet fuel). That hydrogen[35] is currently produced from selected crude oil fractions, natural gas, and coal (limited use). If nuclear power could be used to produce economic hydrogen, it could significantly reduce demand for crude oil and significantly reduce carbon dioxide emissions. The reduction in crude oil demand has important national security, balance of payments, and political implications. In the longer term, assuming hydrogen fueled vehicles are successfully developed, nuclear energy could become the primary source of transport fuels. The growth in hydrogen demand[36] to produce transport fuels is a consequence of two factors: demand for clean fuels and the changing supplies of crude oil (Figure 15). To produce clean fuels, hydrogen is needed to remove impurities such as sulfur, convert carcinogenic compounds in crude oil (such as benzene) into non-carcinogenic fuels, and modify the fuels to burn cleanly. Simultaneously, the quality of crude oils is decreasing as the best crude oils are consumed. High-quality crude oils have a hydrogen-to-carbon ratio near that of gasoline. Low-quality crude oils have a lower hydrogen-to-carbon ratio. Hydrogen must be added to convert low quality crude oils to high quality crude oils and produce transport fuels.

In the future, there may be a demand for carbon-saver fuels. These are liquid fuels where the hydrogen to carbon ratio is maximized to get the maximum energy content per unit of carbon released to the atmosphere. In effect, the liquid hydrocarbon fuel is designed to maximize its use as a hydrogen carrier. It has been estimated that 20% (or higher for low-quality crude oils) of the liquid transport fuel

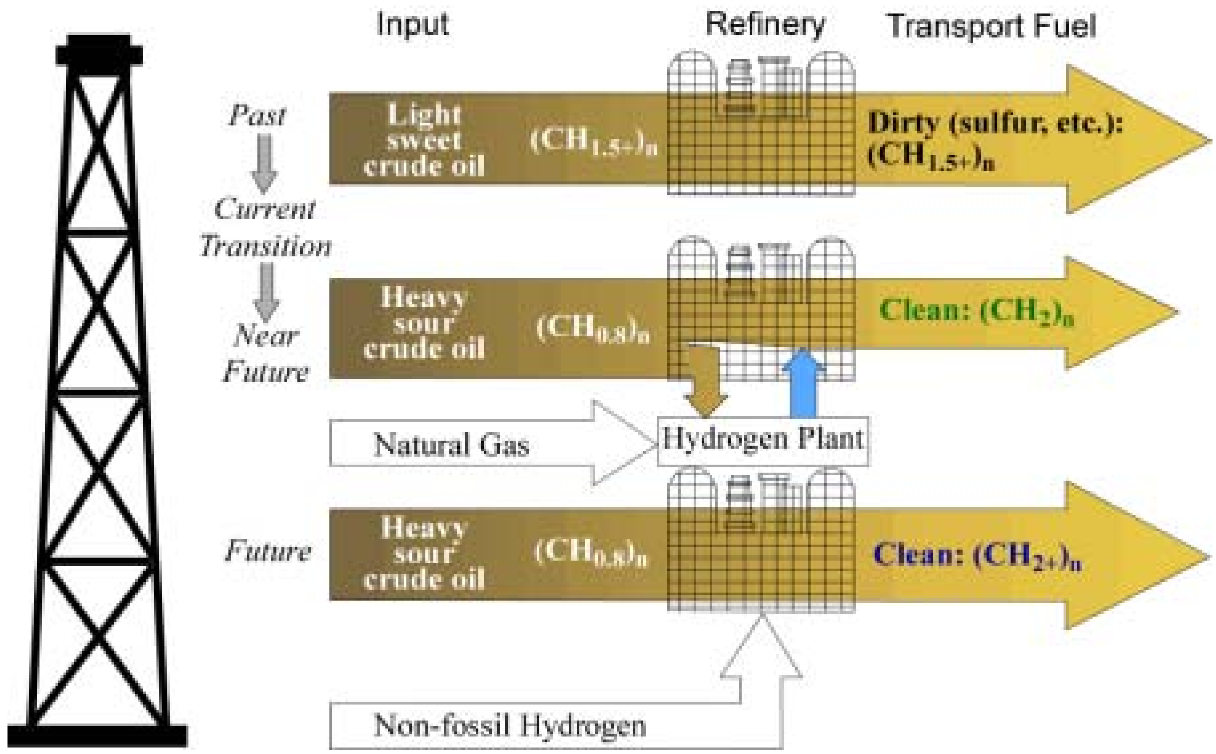


Figure 15. The changing nature of available crude oil supplies is increasing the refinery demand for H_2. (Original drawing provided by ORNL.)

energy content could be provided by maximizing the hydrogen content of the liquid fuels. If nonfossil hydrogen is used, this results in an equivalent reduction in crude oil demand and carbon dioxide releases. Beyond these near-term futures there is the potential for a hydrogen economy where the transport fleet where hydrogen is the fuel. In this context, maximizing the hydrogen content of liquid fuels, with 20% or more of the transport energy indirectly from hydrogen, would create the infrastructure for a transition to a hydrogen economy.

Hydrogen can be produced from nuclear power by thermochemical water splitting.[37] Figure 16 illustrates how heat plus water yields hydrogen and oxygen. Thermochemical processes have potentially higher efficiencies and lower costs than the electrolysis of water with electricity. A typical existing refinery would require a 600-MW_{th} nuclear power plant to produce sufficient H_2 (7.5×10^7 ft^3/day). High temperatures (750–1,000°C) are required for economically viable methods of H_2 production. There are several potential reactor concepts that could meet this application.

Thermochemical production of H_2 imposes a set of technical requirements on the reactor. Firstly, *temperature*: temperatures between 750 and 1,000°C are required. Higher temperatures are preferred. Secondly, *isolation*: heat must be transferred from the nuclear system to the chemical plant at high temperatures. The intermediate heat transport system must be designed to decouple the nuclear reactor heat source and the thermochemical hydrogen production plant such that an accident or upset in one part of the system will not propagate to the other.

Three reactor concepts have been identified that may be compatible with coupling to a thermochemical H_2 production facility. The primary requirement is to provide heat at high temperatures. These are:

- *High-temperature gas-cooled reactor (HTGR)*. Many variants to the HTGR exist, including a pebble-bed reactor and a hexagonal fuel-block reactor.

- *Advanced high-temperature reactor (AHTR)*. This is a modular, molten-salt-cooled reactor that uses a coated-particle graphite-matrix fuel. The AHTR is similar to an HTGR except the high-.

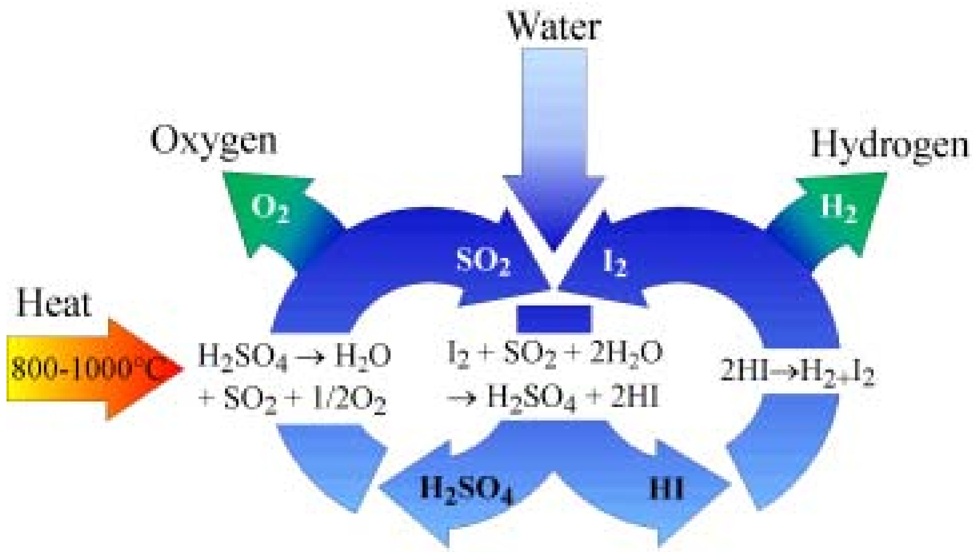


Figure 16. Iodine-Sulfur process for thermochemical production of hydrogen. (Original drawing provided by ORNL.)

pressure helium coolant is replaced with a low-pressure molten salt. Allowable temperatures may be somewhat higher than for the HTGR, and the coolant operates at atmospheric pressure

- *Lead-cooled fast reactor*. This is a lead-cooled, nitride-fuel reactor. The operating temperatures are somewhat lower than those for the HTGR. Lead cooling is required because sodium boils at 883°C—considerably below the required operating temperatures.

There are serious challenges to produce hydrogen using nuclear energy. A further discussion of hydrogen futures and implications is included in Appendix B.

4.3 Modular Deployable Reactors

Number	Concept Name	Sponsorship
–	MMDR —Multi-Modular Deployable Reactors	SNL
–	SPS—Submerged Power Station	INL
–	DORC—Distantly Operated Reactor Complex	IPPE

Commercial nuclear power is primarily considered a baseload generation option, involving large plants, long term planning, extended construction times, and ultimately requiring extended R&D considerations. Power generation applications that require smaller units of capacity, or response times of less than a few years, or that are in locations where the technical infrastructure is not sufficient to support nuclear technology, do not generally include nuclear power as a feasible solution. These types of applications impose new constraints and requirements on the design and implementation of nuclear power that are in addition to the traditional economic, safety, waste, and proliferation considerations of conventional nuclear power approaches.

For these applications, modular deployable reactors provide a path for nuclear power to provide a solution to remote location, shorter response time, and limited maintenance systems. Modular deployable reactors are essentially factory assembled, shipped in single or major units to a site, operated for power generation, and then returned to the supplier as a unit. Modular deployable reactors are designed for installation in locations that are not convenient for conventional construction, or to provide more quickly new or interim power generation capacity than conventional approaches. Offshore underwater deployment is a typical application of such reactors. Modular Deployable Reactors may vary in basic design features but serve the same modular purpose, which is to enable smaller countries, developing countries, or communities with special needs to obtain cleaner, cheaper, rapidly available power.

MDR systems can be based on a range of current reactor technologies, depending on size and application. Although water and gas cooled designs with sizes ranging from a few MW to 500 MW are generally considered, other reactor technologies and sizes can be considered depending on the specific application. Most safety issues are similar to the base reactor technology and the specifics of the application and location. Proliferation and waste issues and fuel cycle options are also based on the base reactor technology, but the modular nature of these systems may provide additional benefits for limiting and monitoring materials.

Although the economics of these systems continues to be of primary importance, the limited alternatives in some of the intended applications may require more flexibility in cost considerations than in traditional applications. The economics of modular deployable systems may benefit more from efficient factory assembly than from economies of scale from large sizes. Modular deployable systems may expand the ranges of nuclear power applications that can be considered in the future. Designs must be tailored to type of application, but the general features of factory assembled, transportable, recoverable systems can provide unique advantages in many specialized applications.

Following is a description of two concepts that make use the economies of manufacturing scale where the overall plant is modular, manufactured in a central factory facility, and can be transported to the generating site. While the output of each reactor is smaller than current plants, the large-scale manufacturing of identical components and their assembly under controlled factory conditions will afford economies greater than the economies of plant size alone.

4.3.1 Multi-Modular Power Reactors (MMPR)

The MMPR concept[38] uses an array of factory-built self-contained power units to form a critical power reactor in a pool configuration. The self-contained modules would be easily transportable by conventional land or sea modes, and would provide power systems in the range of 50 to 300 MWe. The MMPR approach takes advantage of the simplicity, ease of fuel module loading, and large heat sink of a pool type reactor while maintaining the advantages of a factory manufactured system.

4.3.1.1 Multi-Module Description (Gas Cooled Example). The MMPR concept uses self-contained closed-loop modules that are 0.3–1 meter in diameter and 5–12 meters long. Within each module there is a nuclear fuel section and a gas turbine/generator system. The modules are assembled (each module is subcritical so 7–37 together are required for criticality) and operated within a deep-water pool with the necessary reactor controls. Modules could be based on current water-cooled or gas-cooled reactor technologies.

Within gas-cooled modules, the primary coolant (helium) is compressed using the gas turbine compressor, heated in the reactor fuel section, and expanded through a turbine to extract energy. After passing through the turbine, the primary coolant is cooled by passing it along the inside surface of the module. The MMPR primary coolant circulates only within the module. The module assembly is moderated, cooled, and shielded by the deep-water pool.

These modules are then clustered together along with the necessary reactor control system within a deep-water pool that provides the necessary coolant for the closed loop turbine system (Figures 17 and Figure 18). The bulk pool coolant system removes the total steady state power of all the modules, but module cooling at these power levels is accomplished by natural convection along the outside of each module.

Preliminary studies indicate that wide ranges of modular designs are feasible with today's technology, and offer power ranges of 1 to 5 MWe per module. With higher thermal efficiencies, power ranges to 15 MWe per module appear to be feasible.

4.3.1.2 Generation IV Goals—Capabilities of MMPR and Transportability Features.

4.3.1.2.1 Sustainability Aspects—Considering that MMPRs would be deployed in remote locations it would seem that efficient fuel use and low waste features would be paramount. Specific sustainability goals are not discussed here and can be deferred to the particular choice of reactor/fuel type to be deployed with a given MMPR design.

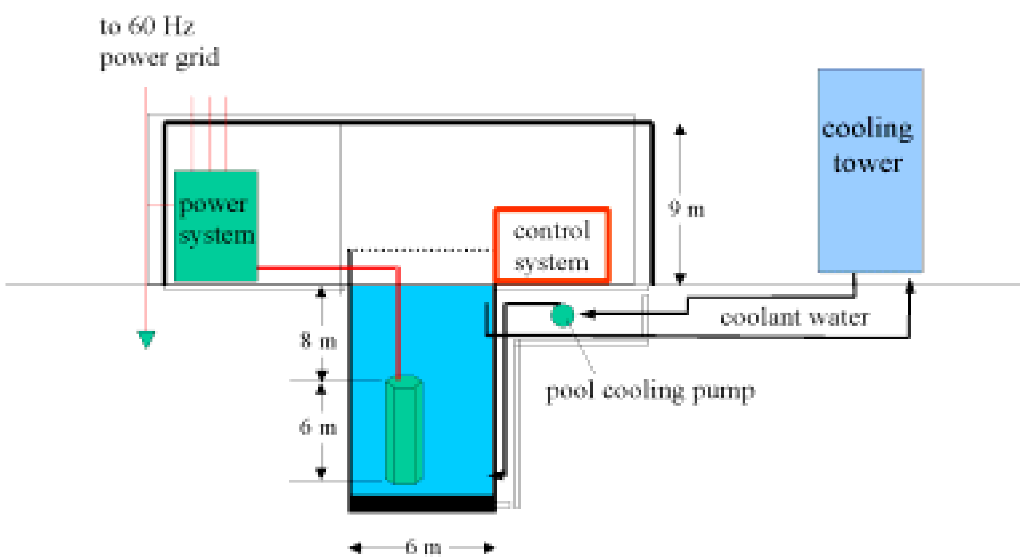


Figure 17. Multi-module power plant layout. (Drawing courtesy of SNL.)

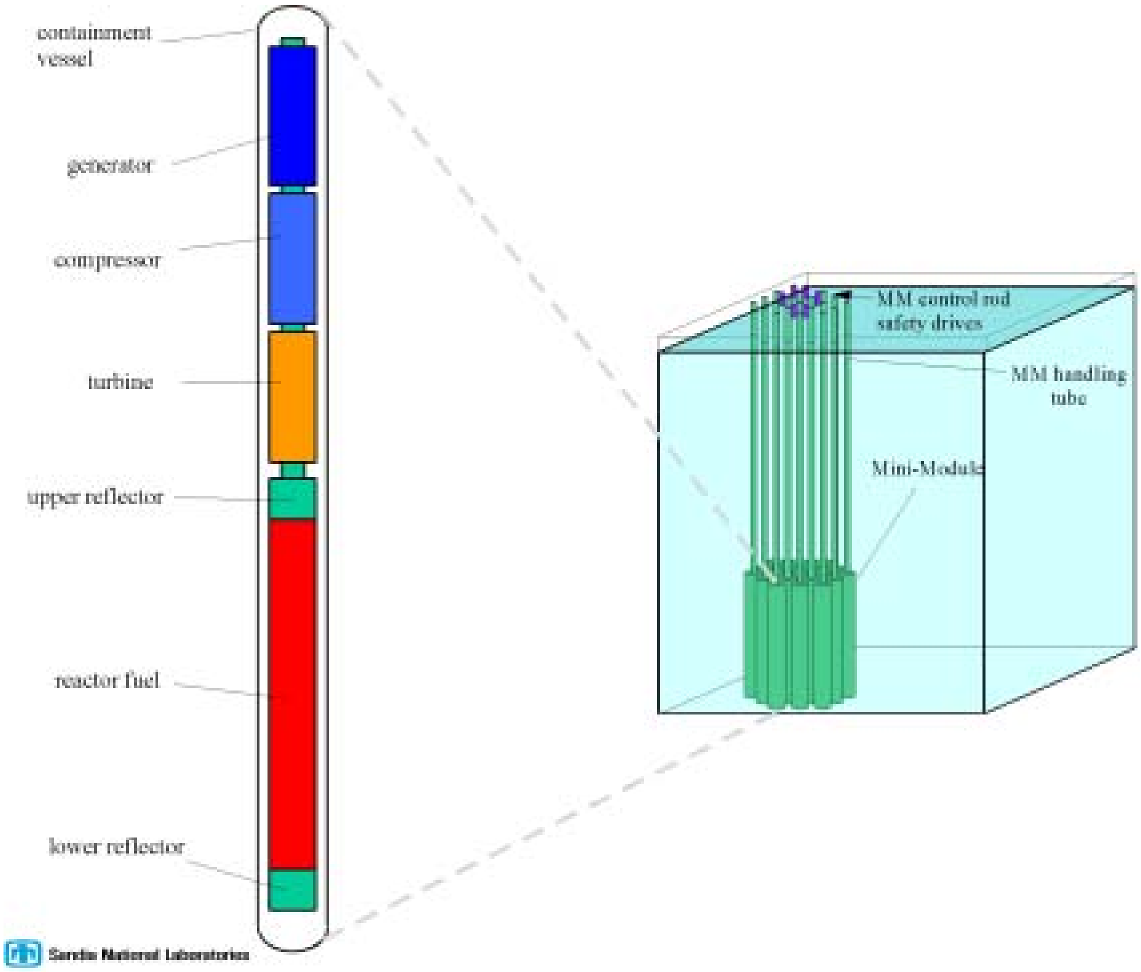


Figure 18. Self-contained multi-module assembly (left), and stack of multi-module assembly units in coolant pool (right). (Drawing courtesy of SNL.)

4.3.1.2.2 Safety and Reliability Aspects—Passive removal of decay heat to the pool heat sink eliminates the problems with pumped cooling. Accident source terms are limited due to the multiple independent modules and the fission product retention characteristics of deep pools. Natural convection or conduction within the module provides sufficient heat transfer to the boundary to limit the temperature rise of the fuel to less than 1,000^{o}C for decay heat removal, with only natural convection within the module. Inherent characteristics would minimize the possibility of fuel damage resulting from loss of primary coolant flow. Operation of the multi-module reactor within a deep-water pool minimizes the impact of a hypothetical failure of a module. If a release is postulated the pool scrubbing action will limit the release to the environment.

4.3.1.2.3 Economic Aspects—A direct-cycle, gas-cooled Brayton system is potentially efficient and economic to implement. Onsite construction would be dramatically simplified, and should be accomplished in a relatively short time—1 to 2 years. Factory built and tested modules could be shipped by truck or rail to the site, loaded, and started up in a relatively short time. For systems in the 10 to a few 100 MW power range, this approach could dramatically change the time and cost to construct new power plants, allowing nuclear systems to be implemented in a similar timeframe and cost to current natural gas plants.

4.3.1.2.4 Transportability Features—The MMPR concept will readily permit the transportation of the nuclear components of the reactor using existing transportation technologies both as new modules and as high burn-up modules. These modules are expected to weigh approximately one to a few metric tons each and will require shielding similar to current spent fuel bundle shipments.[39]

4.3.2 Transportable Submerged Power Station

The second approach is the transportable Submerged Power Station (SPS).[40] This concept is a self-contained unit providing 600 MWe, and is about 20.62 meters in diameter and 115 meters long. The power station is submerged 30 to 100 meters below the surface 10 to 30 km offshore and supplies power to the shoreline by underwater cables. The power stations would be manufactured in a single complex that manufactures the hulls, and installs the reactors and balance of plant equipment. The assembly complex could also be the base for maintenance and refueling of the stations and monitoring of the deployed units.

Figure 19 shows the power station anchored underwater with the power cables taking power to the shoreline in the distance.

4.3.2.1 Generation IV Goals—Capabilities of Submerged Power Station Concept.

4.3.2.1.1 Safety and Reliability Aspects—The power station would be submerged to a depth where wave action and surface shipping present no risk to operation. The SPS is isolated from earthquakes since it is supported above the seabed on pneumatic struts. The station is designed to take advantage of the surrounding mass of water for passive cooling in the event of on-board operating transients or accidents without releasing contaminants to the sea. Operating environments for the submerged power plant appear to be greatly enhanced relative to land based systems due to the separation from atmospheric disturbances and seismic effects.

4.3.2.1.2 Economic Aspects—Due to availability of most key components, construction of the submersible hull/containment for the relatively shallow deployment depths is much less technically challenging than the construction of current military submarines.

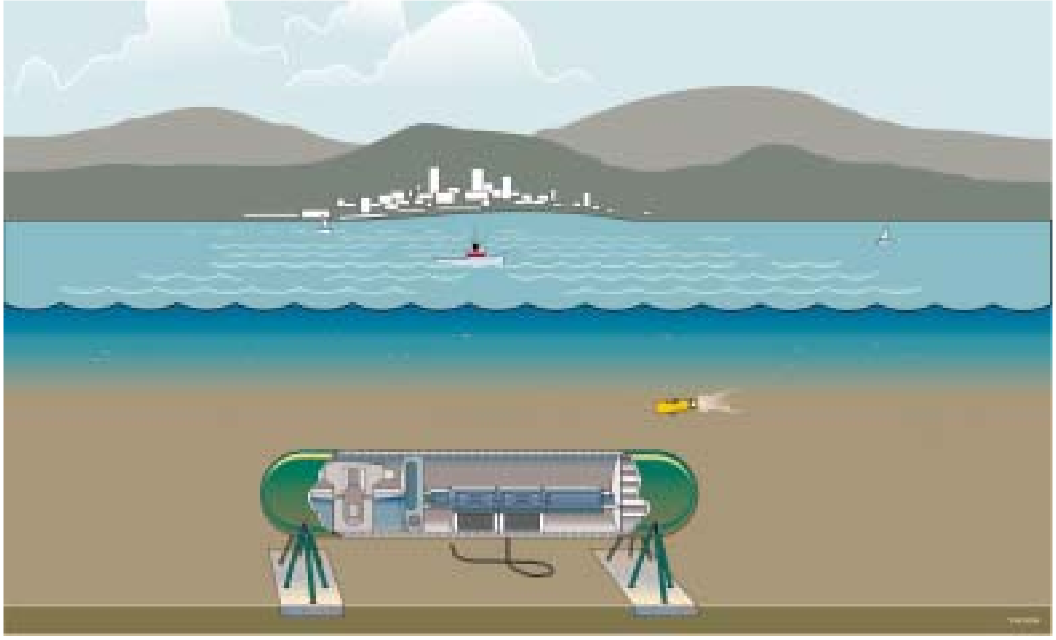


Figure 19. Submerged Power Station concept. (Drawing courtesy of INL.)

4.3.3 Distantly Operated Reactor Complex (DORC)

The DORC program[41] was originally launched as part of the well-known Russian "White Land" program to provide power in remote regions. Small reactors of 10 to 100 MWe can meet the steady growth in demand for electricity and replace existing fossil burning plants. These systems can be used in a number of regions, small towns, and other remote locations.

4.3.3.1 Transportability and Feasibility. Preliminary assessments indicate that with large scale production, such reactors will be inexpensive and will have a high degree of flexibility for siting. The components of DORC-type nuclear power plants can be delivered to site, even by truck or helicopter as need. The necessary technologies which are anticipated to be used for small and midsize nuclear power plants include liquid metal systems and Carnot type gas-steam converters with efficiencies of about 45% (mid level temperatures, steel materials) and up to 50–60% for high temperature applications requiring new materials.

Reactors similar to FSGT and FLGT are envisioned for these applications. These are liquid metal systems with an integral vessel, a single wall separating the primary and secondary system, some arrangements for improving the temperature distribution, and other features. The main characteristics are the same as in the FSGT and FLGT designs described earlier, but there are some additional advantages for the small power plants due to scaling factors.

The DORC program assumes that the distributed system of reactors will be remotely operated from the "base" of a regional service company. The system best suited for this application is a liquid metal reactor with a gas turbine for energy conversions. This system can best follow transients in the operational load. The small inertia of the reactor and the gas turbine is well suited to this requirement. In addition, the high temperatures possible through the liquid metal coolant will result in efficient energy conversion. Therefore either the FSGT or FLGT technologies described earlier in this report are suitable for DORC applications. The IPPE laboratory has developed technical solutions for these situations that allow for minimal reduction in turbine efficiencies for the small plant variant in comparison for larger central power station designs.

The DORC-type plant operates efficiently in a range of 10 to 100% of designed power. In a partial operational mode, there is expected to be a slight increase of 0.5 to 1.0% in generation efficiency. There the DORC-type plant will be economical for both base load and power peaking operation. It can also operate as part of a large scale grid or serving isolated consumers or communities in the so-called "farm reactor" mode. The DORC-type system can generate low temperature heat for civilian communities or provide high temperatures for industrial customers.

4.3.3.2 Generation IV Goals—Capabilities of the DORC Concept.

4.3.3.2.1 Sustainability Aspects—_Sustainability 1_—The IPPE laboratory has developed technical solutions allowing for minimal reduction in power plant conversion efficiency compared to large plants of the same type. Fuel recharging is not required during the 12 to 15 years of operation. The DORC program envisions systems of either remotely operated or fully automated reactors, which are overseen by a regional service company or "base." The fast spectrum reactor and gas turbine both have small systemic inertia allowing for flexible operation and high availability factors. The high temperature liquid metal assures high conversion efficiency. Therefore the FSGT or FLGT technologies are well suited for DORC applications and are seen to have an advantage due to the considerable developmental work that has already been done on these concepts by IPPE.

Sustainability 2–Similar to White Land systems, the DORC concept envisages a specialized (base) enterprise at which various technologies are concentrated for radioactive waste management.

4.3.3.2.2 Safety and Reliability Aspects—*Safety and Reliability 1*—DORC builds on the well-known "White Land" concept. Several variants are under consideration including different types of plants, capabilities, requirements for the energy use, whether it is new or existing, and the number of plants ranging from 2 to 10 among other features. All of these depend on local requirements. The flexibility in the plant designs provide opportunities for optimizing the safety and reliability aspects. The DORC concept is possible only with very effective, reliable, and safe plants, without in situ refueling, and completely automated for remote operation.

Safety and Reliability goals 2 and 3—The DORC concept is expected to have low core damage potential and mitigated offsite emergency response characteristics that are equal to or better than existing conventional nuclear power plants.

4.3.3.2.3 Economic Aspects—*Economics 1*—The primary motivation for the DORC concept is to provide cost effective electricity to consumers. DORC-type plants can operate very efficiently over a range of powers from 10 to 100% of design values. In a partial operational power mode, a small increase in generation efficiency of 0.5 to 1.0% is expected. As a result, DORC-type plants will be profitable both for base load as well as peak load modes of operation. In addition, these designs can operate either as part of a larger electrical grid or be configured to serve isolated consumers or communities in the so-called "farm reactor" mode. Current small nuclear power plant designs also allow for heat or co-generation applications. DORC-type reactor systems can generate low temperature district heat both for civilian communities or high temperature heat for industrial customers.

Economics 2—A significant decrease of DORC NPP construction cost and kilowatt-hour production cost is expected due to higher efficiency and design simplicity. Every DORC-type plant will be provided with a non-nuclear reserve capacity that will produce electricity at the same efficiency of 55 to 60%. Significant work has already been undertaken at IPPE in Russia that will serve to reduce risk in bringing this concept to commercial reality.

4.3.3.3 Research and Development Challenges—DORC. The DORC concept is currently under investigation at the IPPE Lab in Obninsk. Optimization studies include reactor type, plant scheme and layout, transportation, and regional servicing company equipment inventory assessments. Further features requiring ongoing investigation include the use of a multispectrum core, optimization of fuel elements (fuel element type, form, geometry), and maximum use of passive safety principles. This project is currently being carried out in collaboration with the Kurchatov Institute in Moscow. It is hoped that this effort will be extended to include international partners and cooperation on further work on the DORC concept is invited.

4.3.4 Concept Viability Evaluation for Modular Deployable Reactors

The following qualitative assessment in Table 21 is given for modular deployable reactor concepts as a whole group. Lack of specific reactor details make it difficult to provide individual complete score sheets for these concepts at this time.

At this stage the Generation IV evaluation outcome is still to be determined for this concept set. The TWG-4 will temporarily retain the modular deployable reactor set of concepts for further Generation IV Roadmap consideration.

Table 21. Qualitative Group Summary Evaluation—Modular Deployable Set.

Generation IV Goals	Qualitative Assessment
Sustainability 1 Fuel Utilization & Impact	• Better than or similar to present LWRs
Sustainability 2 Waste Management	• Regional utility involvement
Sustainability 3 Weapons Proliferation Resistance	• Hard to hijack a submersible plant
Safety and Reliability 1 Worker Safety and Plant Reliability	• Remote operation • Passive heat removal
Safety and Reliability 2 Low Core Damage Accident Likelihood and Rapid Restart	• Highly robust (depending on coolant technology)
Safety and Reliability 3 Mitigation against Offsite Emergency	• Automated operation
Economics 1 Life-Cycle Cost Advantage	• Modularity eases maintenance • Good efficiency in general
Economics 2 Low Financial Risk	• Simple design

4.4 Advanced Fuel Materials

Number	Concept Name	Sponsorship
–	High Temperature Bicarbide and Tricarbide Fuels	Univ. of Florida
–	High Temperature Carbonitride Fuels	LUTCH[c]

4.4.1 Uranium Tricarbide and Carbonitride Fuels

Advanced fuel materials include solid solutions (Figure 20) of uranium and one or more refractory metal carbides or carbonitrides. Examples of such fuel materials include uranium bicarbides (U, Zr)C and (U, Nb) C, uranium tricarbides (U, Zr, Nb)C, (U, Zr, Ta)C, and (U, Zr, Hf)C, and uranium carbonitrides (U, Zr)CN and (U, Ta)CN. These fuel materials feature very high melting points (>3,500 K), very high conductivity (>25 W/m-K), and chemical stability at elevated temperatures.[42,43,44] Mixed carbide fuels with niobium or tantalum in place of zirconium, increase the thermal neutron capture cross-sections for these fuels and they are intended exclusively for use in epithermal or fast spectrum reactors. Replacing uranium dioxide or uranium carbide with mixed uranium carbides or carbonitrides in any reactor design will result in significant reduction in average fuel temperature and residual heat content, which improves fission product retention and decreases the probability of core melt accident.

c. Scientific Research Associates LUTCH, Podolsk, Russia.

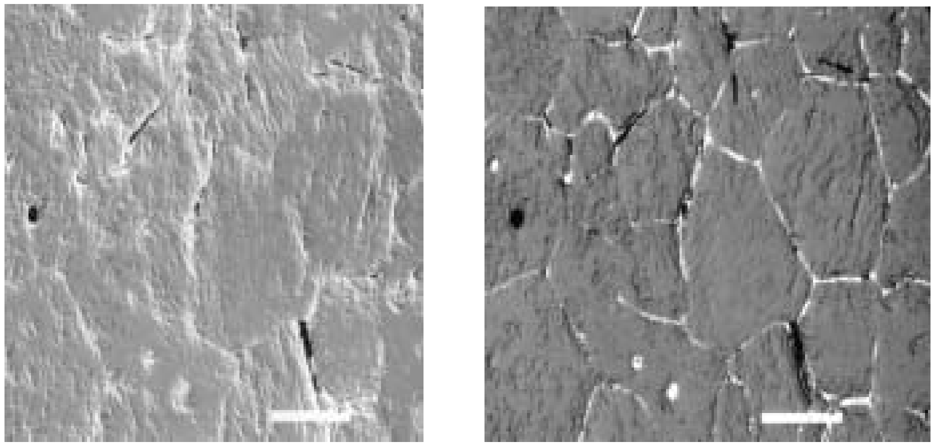


Figure 20. Hypostoichiometric tricarbides, (U0.1, Zr0.77, Nb0.13) C0.95 sintered for 4 min. at 2,800 K (left) and 128 min. at 2,500 K (right). SEM composition contrast in the right image shows up the high compaction. (Scale: 0.5cm ≈ 100µm.)

4.4.1.1 Square Lattice Honeycomb Fuel Wafer Description. A beauty of the fabrication technique shown in Figure 21 is that the stepwise construction allows an end fuel cell with excellent flow and heat transfer properties to be built from simple-to-fabricate modular wafers. A complete fuel cell as shown in the last step would be impossible to manufacture from the raw uranium tricarbide composite material because of the extreme difficulties in extruding solid solution binary carbide fuel elements.[45] The wafer dimensions can be selected so that a specifically desired heat transfer rate can be obtained for constant coolant mass flow rate, by simply varying the size and number of holes in the wafer in such a way as to keep the total flow area constant.

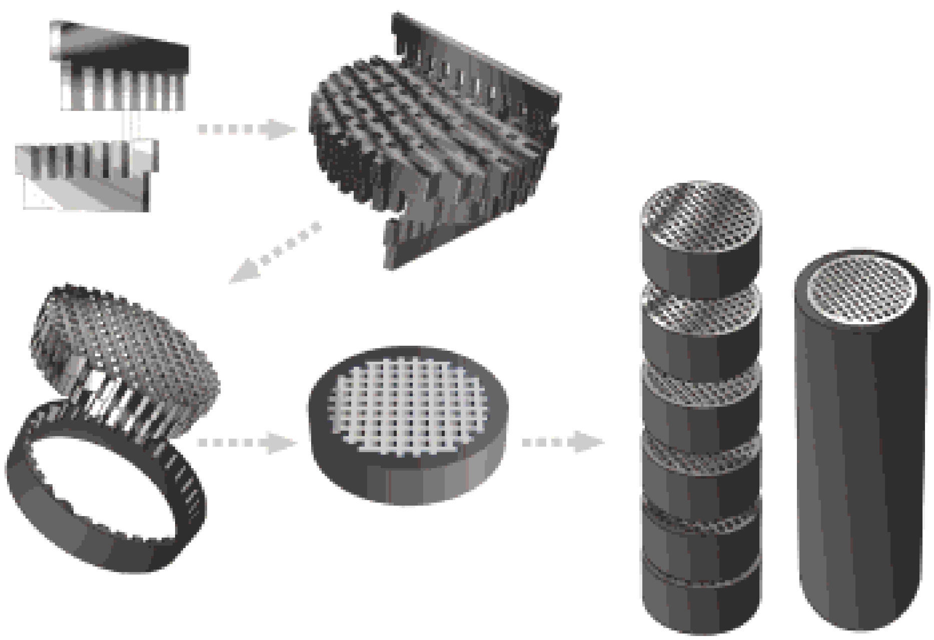


Figure 21. Steps in the fabrication of Square Lattice Honeycomb fuel elements from tricarbide material wafers.

The chief physical characteristics of these mixed tricarbide nuclear fuels compared to conventional oxide nuclear fuels, or nitrides and monocarbides, include: significantly higher melting points (~3,800 K), significantly higher thermal conductivity (~25 to 70 W/m K versus about 3 W/m K for uranium dioxide), and greater stability.

4.4.1.2 Uranium Zirconium Carbonitrides. Uranium refractory metal carbonitrides are developed in the Research Institute of Scientific Associates LUTCH for space power and propulsion applications. In particular (U, Zr)CN has gone through extensive testing and qualification process. The main features of these fuels are very high melting point (~3,600 K) and low dissociation pressure compared with uranium nitride. In essence in these fuels uranium nitride is stabilized at elevated temperatures by forming its solid solution with zirconium carbide.[46,47,48] Despite their attractive features little is known about properties of these fuels and their compatibility with reactor coolants.

4.4.1.3 Generation IV Goals—Capabilities for Tricarbide Fuels.

4.4.1.3.1 Sustainability Aspects—Higher burn-up is possible leading to fewer outages and associated costs for fuel replacement.

4.4.1.3.2 Safety Aspects—The advanced fuel materials afford the possibilities of eliminating or at least significantly reducing core meltdown probability, reducing emergency core cooling requirements due to the higher melting point, significantly increasing the thermal conductivity of the fuel, and significantly lowering the amount of energy stored in the core. With the advanced fuels less thermal energy is stored in the core as a result of the higher fuel thermal conductivity and there is reduced probability of fuel melt or fuel pellet and clad mechanical interaction in an accident scenario (the higher thermal conductivity reduces the fuel centerline temperature and the higher fuel melting point both give an increased margin to fuel melting). The lower coefficient of thermal expansion of the advanced materials reduces the probability of partial or full fuel pellet contact with clad. These tricarbide fuel materials provide characteristics that can significantly improve safety margins for many core designs.

4.4.1.3.3 Economic Aspects—Larger diameter fuel rods are possible reducing fabrication and fueling costs—due to higher thermal conductivity reducing the fuel rod centerline temperature.

4.4.1.4 Research and Development Challenges—Tricarbide Fuels. The tricarbide fuels are developed for space nuclear thermal propulsion applications, where hydrogen coolant is used at about 3,000 K. These fuels still remain to be tested in prototypical conditions, for compatibility with water, and for conditions in high temperature gas cooled reactors. Additional ternary carbides with potentially even further improved thermal properties and chemical corrosion resistance, such as (U, Zr, W)C and the aforementioned (U, Zr, Ta)C and (U, Zr, Hf)C, are in the beginning stages of property testing.

4.5 Alternative Power Conversion Cycles

Number	Concept Name	Sponsorship
N16	Advanced Topping Cycles	Oregon State Univ.
–	REMHD—Radiation Enhanced Magnetohydrodynamics	MIT
N13a	GCR-MHD—Gas Core MHD Cycles	Univ. of Florida

The consideration of additional and new energy conversion technologies is important for three reasons: (1) *New products for Generation IV reactors*: There is an interest in production of hydrogen, fresh water, high temperature process heat, and other special products. Such new considerations may lead to innovative ways to capture the heat and convert it to electricity at different steps in the process. (2) *Efficiency and cost*: Some of the direct energy conversion technologies may improve efficiency or reduce costs by converting either the high temperature heat in reactors directly, or by converting the low temperature heat at the bottom the thermodynamic cycle. (3) *Safety*: It may be desirable to consider isolating future nuclear reactors from the "exposed" secondary power generation subsystems.

4.5.1 Outline of Advanced Topping Cycles

Due to its technical maturity and commercial availability, the steam turbine (Rankine) cycle has been used in almost every major nuclear power plant deployed to date. However, the Rankine cycle is not optimum to take advantage of the much higher temperatures that can be achieved with many of the advanced reactor concepts. Alternative power conversion cycles include all nonsteam turbine cycles with a potential for improving the match to the high temperature nuclear heat source. Gas turbine cycles, such as the Brayton cycle and a few topping cycles, including magnetohydrodynamics (MHD), thermionics and thermoelectrics have been proposed to improve the overall energy conversion efficiency of high temperature reactors. These technologies typically aim to add from 5–15% percent to the overall efficiency of the entire power plant. The combined concepts have the potential for upwards of 40 to possibly 60% thermal to electrical energy conversion efficiencies due to the combined nature of the cycles.[49] The nonclassical reactors that this section reviews are potentially well suited as energy sources for new energy conversion technologies. An example where MHD generators could be employed for high temperature extraction of energy is the Gas Core Reactor (reviewed in Section 3.2). An example where an advanced Brayton Gas cycle could be employed is the Molten Salt Cooled Graphite-Matrix AHTR (reviewed in Section 3.3.1). These topping cycle concepts will require detailed design to couple the transfer of heat from the primary to secondary system and will greatly depend upon the application and the temperatures and materials used.

4.5.1.1 AMTEC. The Alkali Metal Thermal to Electric Converter (AMTEC) is a concept for directly converting thermal energy to electricity. AMTEC operates as a thermally regenerative electrochemical cell by expanding sodium through the pressure differential across a sodium beta alumina solid electrolyte (BASE) membrane. While AMTEC technology is still in the relatively early stages of development, laboratory devices have achieved efficiencies as high as 19% and system design studies indicate that efficiencies as high as 30% are achievable in the near term and 35% or more may be possible. AMTEC conceptually can provide all the advantages of a static power system (low vibration, redundancy, no wear) at efficiencies normally achieved only in dynamic systems. Advanced Radioisotope Power System (ARPS) designs using AMTEC have been designed with 27% theoretical cell efficiencies and 23% system efficiencies. Lab experiments with developmental ARPS type cells achieved 16% efficiencies.

4.5.1.2 Thermionic Diodes. Thermionic power conversion operates on the same principle as a radio tube in that electrons "boil off" from the surface of emitter materials, such as tungsten when they are heated to very high temperatures. The electrons pass across a small gap to a collector surface and produce a voltage that can be used to drive a current through the payload and then return back to the emitter. In thermionic reactor concepts typically employed in space applications in the past, the heat from the nuclear fission in the reactor's fuel elements produces the temperatures needed for thermionic emission to occur.

Although many current and past space nuclear reactor power system designs typically require two coolant systems (primary and secondary coolant systems), the thermionic nuclear power reactor needs only a primary system to remove heat from the collector side of the diode. This primary coolant system then can be directly utilized to reject the waste heat. Typical operating temperatures in past thermionic space reactor design were in the range of 900 K to 1,100 K. These temperatures are relatively cool with respect to available liquid metal component technology, but relatively high with respect to heat rejection in space and require more development efforts in the area of high-temperature, long-life materials.

4.5.2 Advanced Cesium Based Radiation Enhanced Magnetohydrodynamic Cycles

Radiation Enhanced Magnetohydrodynamic (REMHD) systems seek to increase the efficiency of MHD systems by enabling the MHD turbine to expand the gas to even lower temperatures, while maintaining sufficient ionization to maintain the conductivity above the critical point.

REMHD works by using the cascade ionization in the working fluid (caused by high-energy radiation) to establish a nonthermal equilibrium electron concentration, thus enhancing the conductivity of the medium. The overall coolant loop of the reactor is designed to maximize the capabilities of the Cs-REMHD system (Figure 22). As it is relatively easy to condense and evaporate liquid metals using heat-pipe like heat exchangers,[50] a Rankine cycle is the obvious choice. The cesium is evaporated into a high-pressure vapor, expanded through a turbine to extract enthalpy as useful work, condensed to a low-pressure liquid, pumped to high pressure and reintroduced into the evaporator.

4.5.3 MHD Power Conversion for Gaseous or Vapor Core Reactors

4.5.3.1 Concept Description. Magnetic turbines convert the reactor working fluid energy volumetrically and use the very high nonequilibrium electron temperature produced by fissioning in the fissile fluid. The ionizing power of the fissioning process is used to allow for strong coupling between the coolant or the flowing fissioning plasma and the applied magnetic field in MHD. MHD generators are the most widely known "magnetic turbines" with no moving parts, where only the electrodes contact the high temperature fluid. This is important because MHD conversion is being considered here for direct energy conversion of the fission power at temperatures above 1,800 K, which is well above the performance limit of the-state-of-the-art turbomachinery.

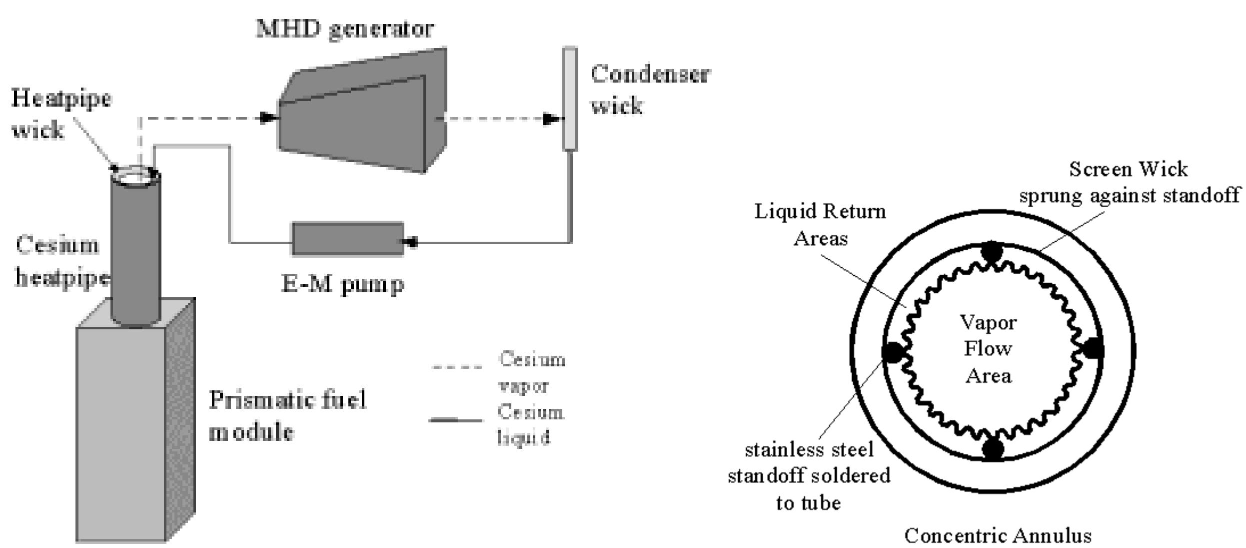


Figure 22. Cesium coolant loop (left), and concentric artery heatpipe (right). (Drawing courtesy of MIT.)

These magnetic "turbines," as illustrated in Figure 23, appear best suited as energy conversion devices for the high temperature fluids produced by the nuclear vapor reactor cores. For these coupled technologies, the material interaction limitations of other energy conversion devices are replaced by a fissioning-conducting vapor interacting with external circuits through electric and magnetic fields. MHD generators are normally DC machines that can operate at modest levels of fluid conductivity (1–100 mho/m), and can be operated in linear or disc flow geometries.

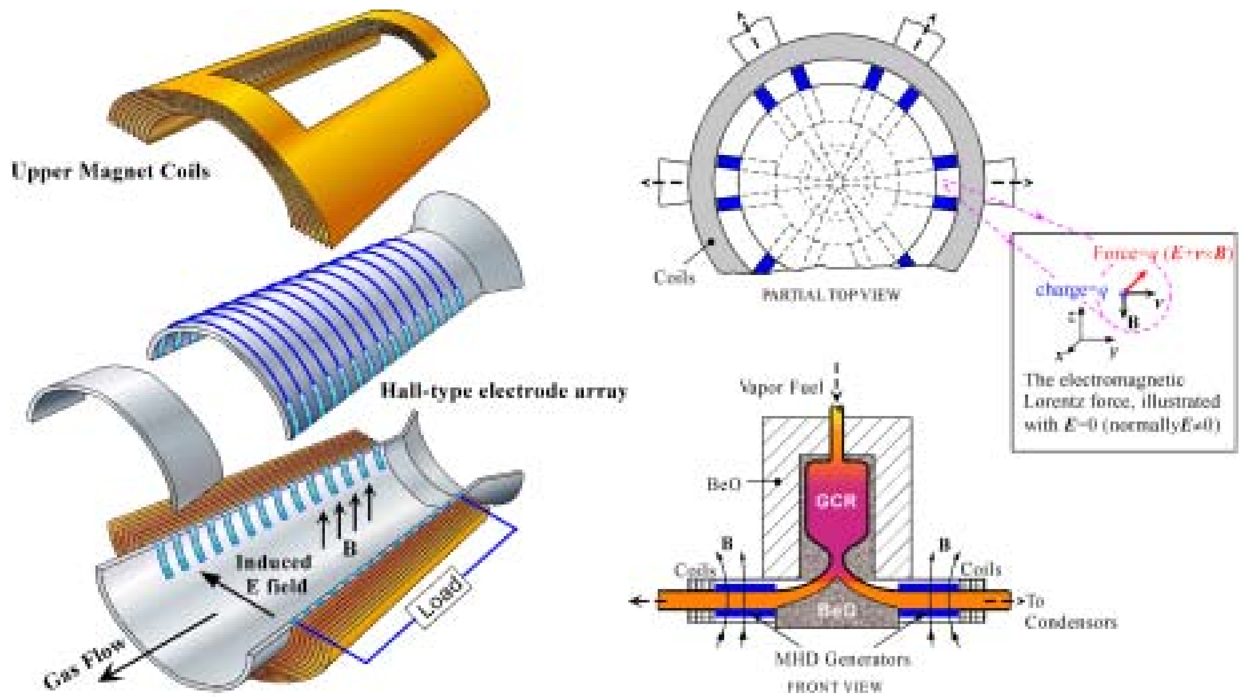


Figure 23. MHD duct configuration, exploded view (left), and schematic for a combined GCR or VCR with an array of line MHD generators (right).

As an example, the class of gas and vapor core reactors (GCRs and VCRs) appear to have special capabilities and opportunities that allow full use of their special fuel and system designs, to achieve very high power conversion efficiencies, by combining power conversion technologies. A power conversion system that appears well suited for coupling with the VCR or GCR (see Section 3.2.1) is one that can use the GCR/VCRs high temperature fluid and its volumetric energy generation, such as an MHD converter, combined with advanced Brayton gas or Rankine heat recovery bottoming cycles.[51,52]

The direct MHD cycle is used to process and convert the fission power between temperatures of 1,800 and 2,500 K. A helium cooled indirect cycle can be used to drive a Brayton cycle that rejects heat to a superheated Rankine steam cycle. The superheated steam Rankine cycle is based on state-of-the-art technology, which is expected to convert an additional 42% of the nuclear energy.

4.5.3.1.1 Disk MHD Generators—Disk MHD generators[53] are especially compatible with the geometry of the gas core reactor. The layout in Figure 24, and the accompanying plot in Figure 25, suggest why. In particular, if the nonequilibrium enhancement by the radiation field and fission fragments proves to be essential to the efficiency of the power conversion process, the integration of the MHD generator with the gas core reactor reflector-moderator system would become a critical design consideration.

A challenging technical issue in the design of a gas core reactor with disk MHD generator is the ability to provide swirl flow that is needed to achieve higher enthalpy extraction efficiency. Parametric analyses presented in Figure 25, illustrate the importance of high swirl numbers in disk MHD devices. A few design options for providing swirl flow in the gas core reactor have been considered and analyzed previously.

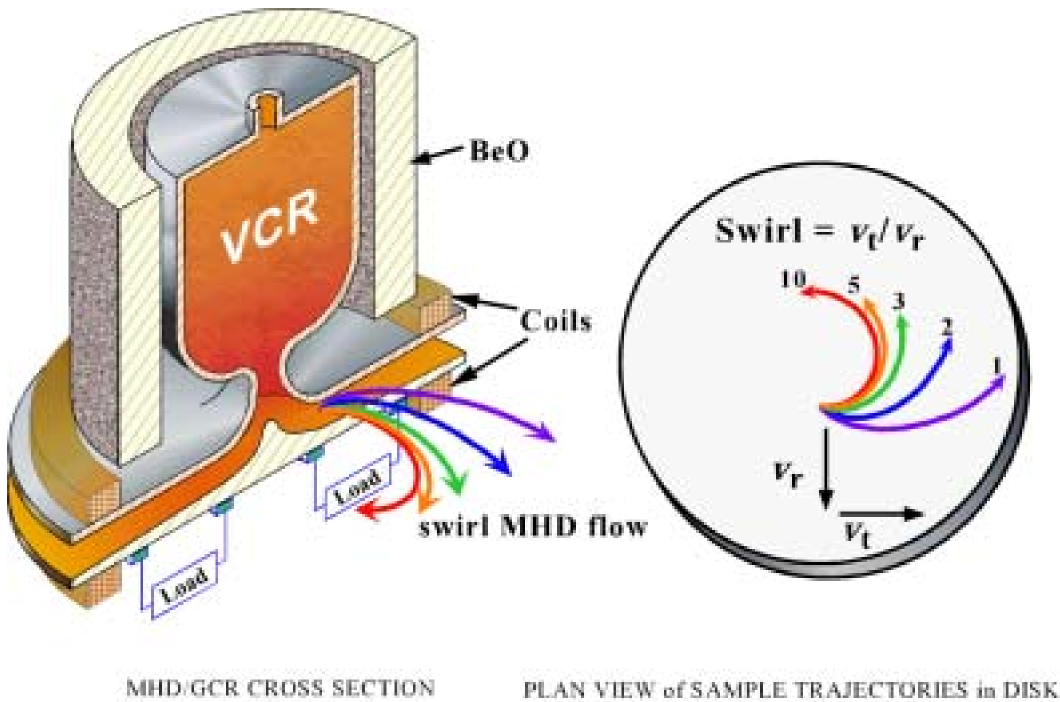


Figure 24. Concept for a disk MHD generator with swirl flow, powered by a gas core reactor.

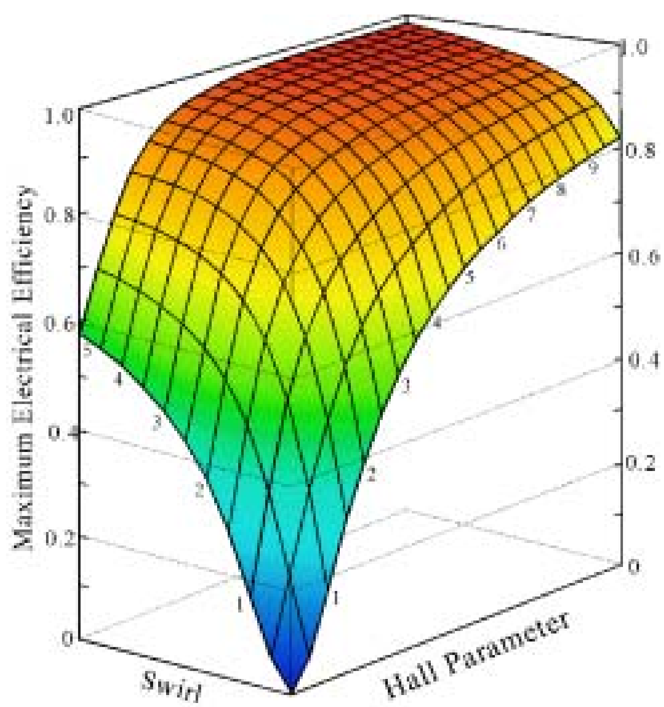


Figure 25. A 3-D plot showing the maximum electrical efficiency as a function of swirl and the Hall parameter. The swirl number = (tangential velocity)/(radial velocity) of the ionized gas in the duct.

4.5.3.1.2 Linear MHD Generators—Compared to a disk MHD generator, a line MHD generator features overtly more design simplicity, although going to higher magnetic fields can introduce complications of electrode arrangement optimization in this type. A number of radially stretched line MHD generators could be used to convert the gas core reactor power to electricity. (See the right-hand side illustration in Figure 23). However, due to the lower power density of line MHD generators, it is not geometrically possible to have the entire length of a line MHD generator within the area of high neutron flux in the gas core reactor system. In particular, the downstream gas temperature in a line MHD generator is not exposed to the high neutron flux that exists in regions closer to the core and reflector/moderator. As a result of lower ionization, the electric conductivity of the lower temperature portion of a line MHD generator may be below what is needed to achieve optimum performance. Yet comparative analysis of the disk versus line MHD generator for coupling with a gas core reactor does not conclusively favor one over the other.

MHD generation has the capability to use the GCR's high temperatures. However, they have very specific technical requirements that impose severe cycle restrictions. The power density in the MHD duct is proportional to the product of conductivity, velocity squared and magnetic field squared, $\sigma v^2 B^2$.

The gas conductivity in turn is proportional to the electron density and the electron mobility. Because high electron density also leads to increased recombination of free electrons with positive ions, it has a negative feedback upon the conductivity, and is therefore undesirable. This is why the aim of MHD power generation is to achieve high electron mobility. Additional conductivity enhancement might be needed from thermal ionization of suitable seed materials, and from nonequilibrium ionization by fission fragments and other ionizing radiation produced by the fissioning process.[54] The need for high electrical conductivity in the ionized gas is one of the limiting factors in the energy conversion efficiency of an MHD generator.

4.5.4 Generation IV Goals—Capabilities of Advanced Power Conversion Cycles

Power cycles in and of themselves do not generally engage the need for sustainability and safety concerns. However, in MHD concepts, because the working fluid is deliberately ionized by fission products this introduces safety issues by increasing the potential volatility of the fuel-gas mixture that is used in the core.

4.5.4.1 Sustainability Features. The power cycles considered here only impinge upon fuel usage and waste to the extent that their parent reactors do. With the potential for higher overall efficiencies that they bring is also the possibility for reducing the need for additional power plants.

4.5.4.2 Safety Features. Additional safety concerns also arise since many of the above power conversion concepts require high temperatures and thus raises the possibility for large amounts of stored energy in the heat capacity of the reactor materials which, if quickly released could cause the distribution of fission products, actinides, and fuel into the operating power plant. These will no doubt need to be specifically considered during the design of any concepts that propose to utilize these high temperature power conversion technologies.

4.5.4.3 Economic Features. Increased economy is the motivating principle for the alternative power cycles. Systems that can be conveniently modularized into individual components such as heatpipes and MHD turbines may find economic advantages. One could imagine constructing graphite moderated prismatic reactors that operate with the core at atmospheric pressure, enabling on-line refueling (by removing the annular core elements and replacing them). Such a reactor could be designed to be significantly smaller than current designs and require a simplified balance of plant.

4.5.5 Research and Development Challenges—Advanced Power Cycles

Overall there are considerable research and development challenges associated with all of the possible alternative power conversion technologies. They will require new materials, may need new fuel forms, and considerable systems design and integration studies to fully become acceptable to the power industry—even for Generation IV concept development. For example, in comparison with coal-fired and conventional partially ionized gas MHD generators the cesium REMHD concept has no experimental basis established, so a major R&D effort must go into modeling and small scale laboratory testing of elements of the concept. Also, while considerable design and development efforts have been applied to thermionic and AMTEC concepts for space application, very little has been done to mate these concepts to terrestrial nuclear system design concepts. Thus, it is not difficult to imagine that a long basic development period is needed prior to considering these advanced power conversion concepts into new reactor systems.

4.6 Other Concepts

There are six further concepts that do not easily fit into any of the above categories, nor do they meet all the requirements of Generation IV strictures. However, due to the informal nature of the advocate reports that were received, these concepts were not immediately forwarded to the Liquid Metal Cooled Technical Working Group, and instead are hereby presented in brief for the sake of completeness (at this stage of the Generation IV roadmap project it has been understood that concepts should not yet be prematurely dropped from consideration merely for scheduling delays). In this section a brief overview of these liquid metal (sodium evaporation) cooled reactors is given.

The remaining four extra concepts are discussed in Appendix C.

The two concepts to be outlined below describe work being done in Russia on liquid metal cooled reactors combined with advanced gas turbines. The first truly successful use of a gas turbine in Brown-Boveri's Velox design laid the foundations for high temperature gas turbines. But temperature levels of 800°C in 1931 were as hard to archive as 1,300°C in the mid-ninetieth. This temperature-limit growth is related to attempts to get the most compact propulsive engine, whether for ship, plane, or train. On the other hand, for power plant turbines, size limitations are not so critical. Some growth in size allows use of specially designed heat exchangers with dramatic efficiency rising without temperature increase.

Nevertheless all existing commercial gas turbines are operating as aircraft turbines do, without any heat exchanger. The fact is that combustion products of fossil-burning plants lead to intensive heat exchanger corrosion and other drawbacks. But for nuclear power plants, this is not a critical factor because the inert gas mixture is not flammable. This means that even with 550°C from any metal-cooled core a properly designed gas turbine efficiency over 40%—greater than Rankine-cycle turbines—can be achieved.

There are a number of successful experiments that have been conducted on small-size prototype gas turbines operating in cycles with internal heat regeneration, and there is wide experience in manufacturing and operating large-size conventional gas turbines. Therefore, it was decided at IPPE to combine this successful data and direct the research efforts towards the mutual adaptations and optimizations of liquid metal cooled reactors and specially designed gas turbine.

The following projects were initiated:

- Fast Sodium-cooled two-circuit reactor with Gas Turbine (FSGT)
- Fast Lead-cooled two-circuit reactor with Gas Turbine (FLGT).

In the Russian national nuclear energy programs, fast neutron spectrum and liquid metal cooled cores are thought to be the safest and most likely prospective basis for future reactors, and the gas turbine component, once perfected, could make them even safer and more competitive.

4.6.1 Sodium Cooled Fast Reactor–Gas Turbine System (FSGT)

The FSGT nuclear power plant project currently includes work on advanced sodium cooled reactors with a gas turbine as the energy converter. Power plants of 600, 800, and 1,600 MWe are under investigation. The FSGT concept is proposed as an alternative to the sodium-cooled systems like BN-1600, Super Phoenix and similar classic ones developed both in Russia and abroad. Unlike steam turbines, the gas turbine efficiency is strongly dependent on gas inlet temperature. That is why the gas turbine based nuclear power plant has a number of innovations for increased operational temperature including power distribution optimization, coolant flow optimization, and others. In many ways, the FSGT nuclear power plant differs from classical sodium reactors. The layout of the core and surrounding structures is shown in Figure 26.

For the purpose of comparisons, it is assumed that the same materials are used for the primary circuit of the FSGT plant and current sodium cooled reactor designs. Even with this limitation, the FSGT achieves heat-to-electricity conversion ratios of ~45% (compared with 40 to 42% efficiency for other nuclear power plants with sodium cooled reactors).

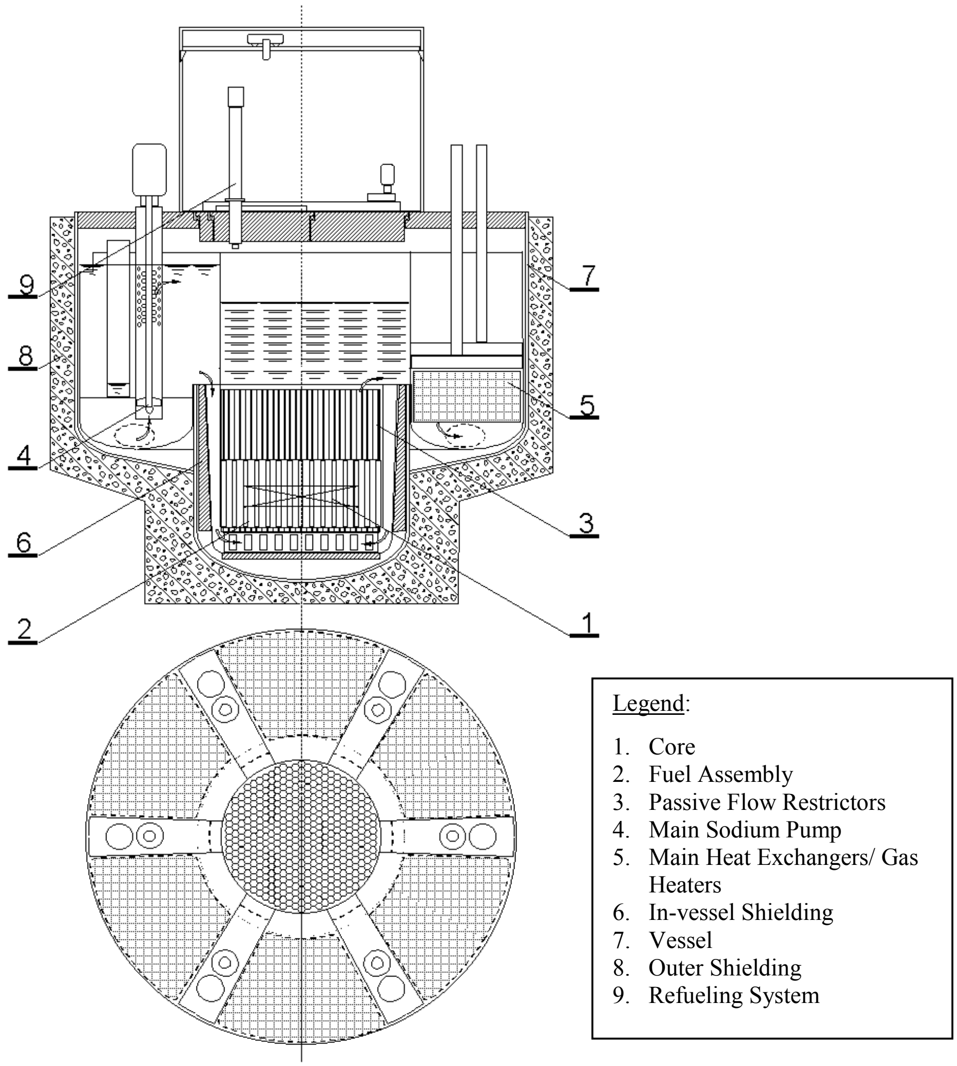


Figure 26. FSGT BN-800 primary circuit layout. (Drawing courtesy of IPPE.)

4.6.1.1 Safety and Reliability Aspects. The combination of a gas turbine and sodium-cooled core solves the "water-to-sodium" problem allowing elimination of intermediate sodium loops. This makes this concept much safer than current designs requiring separate water-liquid sodium heat exchangers. The absence of steam turbines reduces the complexity of the power conversion system, resulting in increased plant availability factors. The inert gas mixture in the FSGT operates at lower pressures than conventional turbines thus increasing plant safety and reliability.

4.6.1.2 Economic Aspects. The elimination of the intermediate sodium loops is a major economic benefit due to increased simplicity and reduction of costs. Current FSGT projects allow heat-to-electricity conversion ratios of ~45% (compared with 40–42% in conventional nuclear power plants with sodium-cooled reactors). Additional efficiency increases result from coolant flow simplification, temperature distribution optimization, heat exchanger volume minimization, construction cost minimization, and other factors.

A distinctive feature of the FSGT gas turbine is a Carnot-type thermodynamic scheme developed at IPPE for nuclear power plant applications. This means that the electricity generation process works close to theoretical limits. Because of the gas turbine design, the conversion ratio remains the same over a wide range of operational power—from 10 to 100% of design power. In a partial operational mode, a small increase of generation efficiency of about 1.5% is expected.

The combination gas turbine and sodium-cooled reactor also allows further increases of the heat-to-electricity conversion ratios up to 60% with the use of high temperature steels and alloys that are compatible with sodium. Preliminary research in this field has already been conducted using the IPPE material database. The intention is to achieve a 50 to 60% efficiency level depending on the materials used and to make some in-core, high-temperature tests in order to prove that these figures can be attained.

The FSGT work includes not only a plant design, but also risk assessment and economic efficiency investigation, including comparison with existing sodium-cooled and water-cooled nuclear power plants and advanced reactor projects of equivalent design power.

International co-operation on further work over FSGT is possible.

4.6.2 Lead-Cooled Fast Reactor–Gas Turbine System (FLGT)

The FLGT project was launched under the V. Orlov initiative. The FLGT project includes work on lead-cooled reactors with gas turbine converters as an alternative to conventional lead-reactor based nuclear power plant projects like BREST-300. Systems with power levels of 300 MW_{th} and 800 MW_{th} are under investigation. The original BREST-300 reactor design has a two-circuit pool-type scheme. The high coolant density has allowed reductions in the dimensions and volumes of the primary circuit. Perhaps the greatest achievement in the FLGT project has been the replacement of steam generators (boilers) by gas heaters without any growth of the primary circuit vessel. At the same time, high coolant density allows some designs to be investigated and tested aimed at increasing the coolant's natural circulation level up to 100% of installed power. If successful, this work will dramatically improve the primary circuit layout. But even now the FLGT nuclear power plant has a number of innovations distinguishing it from the BREST-300.

Experience with lead-bismuth systems has made it possible to manage the impacts of leakage of water into the primary system. The current BREST-300 includes work on minimizing the leakage of emergency water. Meanwhile, incorporating lead coolant and gas turbine technology makes it possible to take advantage of the flexibility of this combination. Therefore, FLGT based nuclear power plants could be safer than BREST plants. The absence of steam turbines reduces the complexity of the power conversion system leading to an increase in the availability factor. The inert gas mixture in the FLGT operates at lower pressure than steam of conventional turbines including the BREST-300 steam turbine, thus increasing plant safety and reliability.

Differences in the coolant properties and primary circuit materials have resulted in considerable changes in the plant layout. The equipment in the primary circuit differs also from the reference case. There is no reliable date for high temperature lead. As a result, the current temperature level in the FLGT design is about 50–61°C lower than the FSGT case. The efficiency in the FLGT design is about 42%, which is similar to the performance of BREST.

4.6.2.1 Research and Development Challenges—FSGT and FLGT Concepts. As with the FSGT concept, the FLGT concept has the potential for further increases in efficiencies up to 60% when high temperature materials are used. However, materials that are compatible with lead are different than materials suitable for sodium systems, so further tests must be made.

Increases in turbine performance are not the only factors for better utilization of nuclear energy. Some 30–70% of the energy can be lost during transmission, and other losses occur in poorly designed transformers. There have been a number of proposals to site reactors closer to consumers. The PRISM and PIUS designs are the best known. However, these were meant to be large scale plants, and as a result may not be best suited for this purpose. Systems that can effectively follow changes in the electrical demand are needed, so PRISM, with a steam turbine cycle and PIUS that operates with a thermal neutron spectrum, could not meet the plant dynamics requirement. Similar limitations exist for plants with thick pressure vessels, which eliminates consideration of the gas cooled and vapor cooled designs. As a result, the FSGT and FLGT technologies where selected for the Distantly Operated Reactor Complex, which was described under "Modular Deployable Reactors," in Section 4.3.3.

5. CONCLUSION

Overall, the TWG-4 has conducted a global search to identify and evaluate new power reactor concepts. The group has also devoted time to reassess some of the older reactor ideas that were not developed further because of research or commercial priorities. A noteworthy feature of the collective effort is the great potential that almost all of the submitted concepts have in being able to meet and occasionally exceed the demands on safety, sustainability, and economic viability required by the Generation IV Reactor type specification. The TWG-4 is pleased with the quality and diversity of the concepts submitted. The submissions came from universities and research laboratories from both inside the U.S. and overseas.

Due to the low level of technology maturity of nonclassical reactor concepts, a head-to-head comparison with the state-of-the-art light water reactors proved to be a daunting task. Often it was necessary to use imagination to envision the level of technological maturity of a concept needed for meaningful discussions on safety, design certification licensing, and economics. With that in mind, the nonclassical concepts are judged based on their unique and innovative design features as well as their potential for achieving the Generation IV sustainability, safety, and/or economy goals. Although qualitative scoring was reached by the majority opinion, the spread in quantitative scoring is rather high. In particular, for reactor concepts with significantly innovative design features the spread of opinion is broader.

The concept sets that passed the initial screening for potential are:

- Molten Salt Core Reactors
- Gas Core Reactors
- Advanced High Temperature Molten Salt (Nonconventional) Cooled Reactor
- Organic Cooled Reactors (also Nonconventional Cooled).

The concepts that did not pass the screening for potential were:

- Solid-State Heatpipe Cooled Reactors (Non-Convection Cooled)
- Direct Energy Conversion Reactors.

There are many research and development gaps, feasibility issues, and safety concerns that do not fall within the objectives of this preliminary report and will be addressed in future R&D scope reports. Hopefully after research and development plans are devised for nonclassical reactor concepts, one would hope that the spread of opinion would shrink.

As it was noted in the discussions, research is needed in high temperature fuels and materials, advanced power conversion cycles, diversification of the energy products, and to some extent in control strategies. It is also fair to conclude that more rigorous analysis of safety implications needs to be made because these may not be the same as those of classical reactors. It is believed that there is insufficient knowledge to fully understand all design implications and to answer all questions regarding safety and economy with a high degree of confidence and certainty. But the research needs are very broad, interesting, and challenging, and indeed may overlap with those of many other classical reactor concepts.

Despite many technology gaps and data uncertainties, there is no lack of innovation and revolutionary ideas in nonclassical reactor concepts. Several concepts such as gas/vapor core reactors meet or exceed Generation IV goals for sustainability, safety, and economy, and have potential for making huge inroads toward achieving the optimum use of nuclear energy.

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Appendix A

Proliferation Resistance of Molten Salt Reactors

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Proliferation Resistance of Molten Salt Reactors

Introduction

The TWG-4 defined the nonproliferation characteristics of a molten salt reactor (MSR) as equivalent to an LWR. There is the potential that the nonproliferation characteristics of a MSR could be significantly better than an LWR; however, the uncertainties are large. As a consequence, the TWG (1) identified nonproliferation characteristics of MSRs similar to an LWR and (2) identified this area as requiring additional research and development. The uncertainties are of two types. First, at this stage of development, there are many design options and it is not clear what designs will prove most practical. The design details can have a major impact on the proliferation resistance of any reactor and fuel cycle. Second, research and thinking in nonproliferation has not examined these fuel cycles in any depth.

The early work on MSRs was conducted before nonproliferation was a major policy concern. The experimental reactors operated on a ^{233}U/thorium-fuel cycle with no significant ^{238}U in the fuel; ^{233}U is a weapons-usable material. Recent studies of MSRs have included variable quantities of ^{238}U to isotopically dilute the ^{233}U.

There has also been limited work on understanding the proliferation characteristics of these reactors and the associated fuel cycles. For example, there is a clearly defined boundary between weapons-usable and nonweapons-usable ^{235}U. If the concentration of ^{235}U in ^{238}U is above 20%, the material is weapons-usable. This allows one to state that LWR fresh fuel does not contain weapons-usable materials. In contrast, there is no formal equivalent definition of the boundary between weapons-usable and nonweapons-usable ^{233}U. Many MSRs use fuel cycles containing mixtures of ^{233}U and ^{238}U. It is difficult to conduct an analysis of proliferation resistance when there is not even a formal definition of weapons-usable ^{233}U. The technical analysis generally assumes that the boundary is ~12wt% ^{233}U in ^{238}U.

This appendix defines the issues associated with proliferation-resistance for a MSR and contrasts the different proliferation-resistance strategies for different reactors and fuel cycles. Much, but not all, of the discussion would also be applicable to other fluid fuel reactors with thermal neutron fluxes that use a ^{233}U/thorium-fuel cycle.

All fuel cycles use some common safeguards strategies. These common strategies are not discussed herein. There are many MSR safeguards strategies. Only some of these will be discussed herein.

There are three basic fuel cycle safeguards strategies.

- *Once-through fuel cycle*. The basis for proliferation resistance is to minimize processing of materials that contain weapons-usable materials. This minimizes the number of commercial facilities that could be modified to produce weapons useable materials. The SNF is directly disposed of. Once-through fuel cycles result in a continuously increasing inventory of weapons-usable materials in SNF. Medium quality plutonium is produced in the SNF from LWRs.

- *Closed (fast reactor) fuel cycle*. The basis for proliferation resistance is (1) security in sensitive facilities and (2) destruction of weapons-usable fissile materials that ultimately go to the repository. The total inventory of weapons-usable fissile materials is capped. The systems can be designed so the weapons-usable materials are limited to a subset of facilities (except for the SNF)

- *MSR fuel cycle*. The basis for proliferation resistance is (1) minimizing the total quantity of weapons-usable fissile materials in the fuel cycle, (2) very-low grade isotopics for weapons-usable materials, (3) no off-reactor-site fuel-cycle facilities (including the repository) with weapons-usable materials, and (4) limited process capability.

Because the philosophy and basis for proliferation resistance is different for each fuel cycle, it is difficult to make cross comparisons. Furthermore, many of the issues associated with MSRs involve complex questions about isotopics and chemistry.

Factors In Proliferation Resistance

Isotopics

The usability of fissile materials in nuclear weapons depends on the isotopics. Fast reactors have significant inventories of high-quality plutonium, LWRs have significant inventories of medium quality plutonium, and MSRs have fissile materials with poor characteristics in terms of use in weapons.

All fissile materials (uranium, plutonium, etc.) must be considered. MSRs are generally designed as breeder reactors that operate on ^{233}U-^{232}Th fuel cycles with variable amounts of ^{238}U. For proliferation resistant systems, sufficient ^{238}U is added to convert the ^{233}U and any ^{235}U to nonweapons-usable uranium.

In a molten salt reactor, the fuel is one homogeneous batch. If a proliferation-resistant reactor is required, this has major restrictions on fuel cycle and reactor design. If the reactor has high conversion ratio or is a breeder reactor with a conversion ratio >1, the feed after reactor startup will be a mixture of thorium, depleted uranium, or low-enriched uranium (LEU). If the reactor is a breeder reactor, there is no need for an enrichment plant after the reactor is initially fueled with LEU. If the conversion ratio is too low, there are two alternative fuel cycles.

- *High-enriched feed*. In a LWR, the fuel is LEU. The ^{235}U is burnt out and the SNF with the excess ^{238}U is removed. One of the primary purposes of refueling an LWR is removal of excess ^{238}U from the reactor. In a fluid fuel reactor, all of the fuel has the exact same composition. There is no way to easily remove the ^{238}U. It is mixed with the ^{235}U. If the conversion ratio is too low, HEU or another concentrated fissile fuel is required as fuel. If LEU is used, the ^{238}U will build up in the machine until it exceeds the solubility of the salt solution and precipitates from the solution.

- *Recycle uranium*. A MSR with a low conversion ratio can be fueled with LEU providing excess uranium from the reactor is removed. However, the fissile content of the uranium removed will be the average fissile content of uranium in the reactor. In an economically practical system, this would require an enrichment plant to recover this fissile uranium.

In a MSR, a breeder reactor is the proliferation resistant reactor. This is exactly opposite of the conventional thinking associated with solid-fuel reactors. If a MSR is a breeder reactor, there are, after providing LEU for startup, no away-from-the-reactor facilities that contain weapons usable materials or that could be converted to make weapons-usable materials; thorium, natural uranium, and depleted uranium are the fuels.

The ^{233}U-^{232}Th fuel cycle has one other unique characteristic. Unless very special production techniques are used, the production of ^{233}U results in secondary production of ^{232}U, which decays through several decay products to thallium-208 (^{208}Tl), which then emits a 2.6- MeV gamma ray. Consequently, the radiation field associated with ^{233}U increases with time. Within months after uranium purification with removal of the decay products, the radiation levels from an 8-kg sphere of reactor grade ^{233}U will be 10s of R/h at 1 meter and hundreds of R/h at 1 ft. Eight kilograms is the IAEA equivalent definition of the quantity of ^{233}U needed to produce a weapon. For highly-irradiated ^{233}U with a high ^{232}U content, these radiation levels become significant barriers against use in nuclear weapons or purification of ^{233}U by isotopic separation from other uranium isotopes. Only isotopic separations can separate the ^{232}U from the ^{233}U and permanently eliminate the radiation hazard. This is a much more difficult separation than separating ^{235}U from natural uranium because of the small difference in mass and the high radioactivity. Note that this proliferation barrier is separate and distinct from isotopic dilution of ^{233}U with ^{238}U.

The addition of ^{238}U to a MSR to isotopically dilute ^{233}U to nonweapons useable uranium results in the production of plutonium. Unlike other fuel cycles, in a MSR the fission products are removed from the salt. The actinides remain in the salt until they are transmuted to higher actinides or fission. This fuel cycle combined with the thermal neutron flux in the reactor results in unusual plutonium isotopics, as shown in Table 1-1.

Table 1-1 shows isotopics for weapons-grade plutonium, PWR plutonium, a proliferation resistant MSR, and a MSR used to destroy actinides from LWR SNF. Fast reactor plutonium would have an isotopic composition between weapons-grade plutonium and PWR plutonium. The IAEA defines the quantity of plutonium necessary to manufacture one weapon as 8 kg. When this definition was developed, power reactors produced plutonium that was 80+% ^{239}Pu. No one was considering a reactor in which ^{239}Pu is a minor plutonium isotope. The critical mass of ^{242}Pu is about an order of magnitude greater than that for ^{239}Pu. Different nuclear isotopes have different nuclear properties. The implication is that for an MSR, the IAEA definition of the quantity of weapons-usable plutonium should be modified to account for plutonium isotopics. If 8 kg of ^{239}Pu were required to build a weapon, a significantly larger, but currently undefined, quantity of plutonium would be required, providing the primary plutonium isotope were ^{242}Pu. The safeguards requirements change with isotopics.

Table 1-1. MSR plutonium isotopics (%).

Isotope	Weapons Grade	Reactor Grade[a] (PWR)	Denatured MSR Breeder[b]	LWR Actinide Recycle in MSR[c]
^{239}Pu	93.	56.6	30	4.5
^{240}Pu	6.5	23.2	18	17.9
^{241}Pu	0.5	13.9	14	5.0
^{242}Pu	0.0	4.7	38	70.2

a. PWR SNF also has 1.3% ^{238}Pu.

b. Engel 1978.

c. Includes 2.3% ^{238}Pu (Greenspan 2001); isotopics are dependent upon choice of coolant and operating temperature that control solubility limits.

Quantities

The inventory of weapons-usable materials in an MSR that uses isotopically diluted ^{233}U is very low. For any particular reactor, there will be a ratio of weapons-usable fissile material to nonweapons-usable fissile material (plutonium to isotopically diluted ^{233}U). If the plutonium inventory is to be minimized to meet the proliferation-resistance objectives, the nonweapons-usable fissile inventory must also be minimized. Fast reactors typically have 10 times the fissile inventory of that of thermal reactors per unit of power production because of the low absorption cross sections at high neutron energies. For example, a 1,000-MW(e) LWR with thermal neutron spectra has about 3 tons of ^{235}U and ^{239}Pu in its reactor core. A MSR has about half this inventory. In contrast, an equivalent fast reactor has 25 to 35 tons of ^{235}U and ^{239}Pu.

Preliminary assessments indicate that the ratio of plutonium to ^{233}U in such a reactor would be expected to be between 0.01 and 0.02. This is a consequence of several synergistic nuclear effects.

- *Plutonium generation*. In a thermal neutron spectrum, fertile materials, such as ^{238}U and ^{232}Th, have low neutron-capture cross sections, whereas the fissile materials, such as ^{233}U and ^{239}Pu, have high fission cross sections. In a ^{233}U-thorium reactor, there are large quantities of thorium, usually 40 to 60 times as much ^{232}Th as ^{233}U. What this implies is that most of the neutrons are absorbed in ^{233}U and ^{232}Th. The ^{233}U absorbs many neutrons because of its large cross section. The ^{232}Th absorbs many neutrons because there is so much of it. In contrast, the ^{238}U has a low nuclear cross section, and there is not much of it in the reactor, only enough for isotopic dilution of the ^{233}U. The ^{238}U does not absorb many neutrons; therefore little plutonium is made.

- *Plutonium destruction*. The nuclear cross section of ^{239}Pu is significantly larger than that of ^{233}U. Plutonium-239 is preferentially destroyed in the reactor.

The IAEA defines 8 kg of plutonium as the quantity necessary to construct a weapon based on the historic assumption that all plutonium is primarily ^{239}Pu. If one accounts for the unusual plutonium isotopics, a few tens of kilograms of MSR plutonium would be equivalent to 8 kg of ^{239}Pu. This, in turn implies that the total quantity of weapons-usable plutonium in such a MSR would be near that to construct a single weapon. That plutonium is diluted in about 100 tons of fuel salt. In comparison, the inventories of weapons-usable plutonium in LWRs are equivalent to many tens of weapons.

External Facilities

Except for initial startup, it may be feasible that no facilities external to the reactor and associated process operations contain weapons-usable fissile materials or even concentrations of fissile material beyond that of natural uranium. This includes the repository. The thorium fuel cycle minimizes the generation of higher actinides. In the normal mode of operation, fission products are removed from the molten salt but actinides remain until they are fissioned. The current estimates for the French AMSTER concept indicate a deposit of about 20 grams of transuranium elements in the wastes to the repository for each TWhr produced, compared to a deposit of 30 kg per TWhr for an LWR with an average burn-up of 60 GWd/ton.

Proliferation Analysis

In the normal operation of a MSR and associated fuel cycle with isotopically diluted ^{233}U, (1) there is no SNF, (2) the inventory of weapons-usable fissile materials is very low in the reactor and associated process facilities, and (3) the concentration of weapons-usable fissile material is very low in the ~100 tons of salt. This has two implications.

- *Terrorist threat*. There is no terrorist threat for stealing weapons-usable materials because there is no significant inventory of weapons-usable fissile material at the reactor or in the fuel cycle.

- *National (governmental) threat*. For the nation state that wishes to divert weapons-useable materials, there is very limited inventory of weapons-usable fissile material. One cannot build in secret a large capability to recycle SNF, divert the SNF, and shortly afterwards have a significant inventory of weapons-usable materials.

Proliferation risks associated with a MSR involve modification of the reactor and associated facilities for the production of weapons-usable materials. There are several options including (1) stripping out plutonium as it is made to avoid unacceptable isotopics, (2) intercepting $^{233}Pa/^{232}Pa$ before they decay to $^{233}U/^{232}U$ and are isotopically diluted with ^{238}U, and (3) other routes. However, there are several complications in modifying the reactor.

- *Fissile quantities.* The breeding ratio of the reactor is low. When ^{238}U is added, it is only a little above 1. There is little excess fissile material in the reactor and associated facilities. Large-scale removal of fissile material will shut down the reactor with the loss of power generation.

- *Detection*. If the reactor were safeguarded, it would be expected that physical modifications to the facilities would be quickly detected. Furthermore, any changes in operation will result in changes in the composition of the fuel salt. The homogeneous fuel salt simplifies monitoring since there is only one composition in the reactor—not wide variations in composition as with solid fuel assemblies.

- *Choice of fissile*. In a solid fuel reactor, one can place targets to produce weapons-usable fissile materials in a fuel assembly. The target material and irradiation conditions can be chosen. The reactor can be converted from one mode of operation to another without major plant modifications. This is not an option for a MSR. The molten fuel implies a homogeneous composition of the fuel.

There are two important caveats in this analysis.

- *Design goals*. MSRs can be built in many configurations. The proliferation-resistance is dependent upon the configuration. At the current time, there do not appear to be large penalties associated with designs that minimize proliferation risks. However, the early state of research in this area leaves many questions. In this context, the early work on the MSR was before proliferation-resistance was a major issue.

- *Proliferation analysis*. There has not been a major effort to understand the proliferation resistance of such very unusual fuel cycles. Until more work has been done, there will be limited confidence in the conclusions.

Conclusions

MSRs using isotopically diluted ^{233}U have radically different characteristics in terms of proliferation resistance compared to open-fuel-cycle LWRs or closed-fuel-cycle fast reactors. There is the potential for reactors and fuel cycles with much better proliferation resistance than LWR open fuel cycles. Significant work is required to understand the proliferation characteristics of such fuel cycles.

Appendix B

Nuclear Power and Hydrogen

Appendix B

Nuclear Power and Hydrogen

This appendix includes the information from an article published in Nuclear News, as well as material from an earlier report of the Nonclassical Working Group, and some additional detailed information. There are two items that deserve special note. First, less than half the energy worldwide is used to make electricity. If the world energy supply problem is to be addressed, the nonelectrical half of the energy demand must also be addressed. Transportation is the biggest component of this other half. Second, the appendix adds information on the potential to push the hydrogen content in liquid transport fuels such that 20% of the energy value of the fuel would be from the added hydrogen. Boosting the hydrogen content of liquid fuels has several benefits: (1) reduced carbon dioxide emissions, (2) reduced imports of oil and natural gas, (3) method for nuclear power to make a significant contribution to the transport energy demands, and (4) easy transition to a hydrogen economy.

Introduction

A worldwide reevaluation of nuclear energy is under way. One of the most fundamental questions not typically being asked, is: “What should nuclear energy be used for?” Historically it has been assumed that nuclear energy will be used primarily to produce electricity and that other applications are rather limited. However, electricity meets less than half the world’s energy needs. Very large quantities of energy are required for transportation, chemicals, and production of materials such as steel. Many, perhaps most, of these nonelectrical needs can be met using hydrogen (H_2). If the H_2 is produced using nuclear energy, greenhouse impacts from fossil fuels would be reduced and imports of crude oil would be reduced. The long-term market for H_2 may ultimately equal that for electricity. The near-term markets for H_2, long-term markets for H_2, transitions to H_2 futures, methods to produce H_2 from nuclear energy, and related issues are discussed herein.

Current and Near-Term Trends

The world[1] consumes about 50 million tons of H_2 per year. If this H_2 were burned, energy would be released at a rate of 200 GW. Most of the H_2 is made from natural gas.

In 1999, Ogden[2] estimated that the energy value of all the H_2 consumed in the United States was about 1.5×10^{18} J/year (50 GW). Almost all of this H_2 is used for chemical and refinery purposes. In the United States, H_2 production has been projected to grow by as much as a factor of four to 6×10^{18} J/year (200 GW) by the year 2010. Most of the increased demand is for H_2 by refineries to convert lower-grade crude oils into clean gasoline.

If all the H_2 in the United States in 2010 were produced using nuclear power and the thermal-to-H_2 conversion efficiency were 50%, the thermal energy required would exceed the current energy output of all the nuclear reactors that exist in the United States today. In the longer-term, if the United States moves toward the popular concept of a “hydrogen economy,” the demand for H_2 would increase by one to two orders of magnitude beyond these H_2 consumption rates, with the energy requirements for H_2 production exceeding those for electric production.

Why the Growth in H_2 Demand?

Hydrogen currently has three principal uses in industrial processes: fertilizer production, manufacture of chemicals, and refining of crude oil stocks. Historically, the chemical industry has been the largest consumer of H_2. Hydrogen is the primary input to manufacture ammonia (the principal fertilizer used worldwide). Fertilizer production facilities continue to be a major consumer of H_2 today; however, the market is not expected to grow rapidly. Hydrogen is also used in the production of many other chemicals (such as methanol) and in some parts of the world for conversion of iron ore to metallic iron. Most of the H_2 used in these processes is manufactured from natural gas.

The most rapidly growing component of H_2 demand relates to crude oil refining. The H_2 production capacity of the world's refineries[3] is 1.15×10^{10} std ft^3/d (46 GW), with a United States refinery hydrogen production capacity of 3.56×10^9 std ft^3/d. Hydrogen is primarily used to convert heavy crude oils into gasoline, diesel, and jet fuels. A combination of factors is rapidly increasing demands for H_2. The world is exhausting its supplies of high-quality crude oils and is thus using lower quality heavy crude oils. There is also a demand for higher performance, clean fuels. Furthermore, the demand for heating oils is decreasing, while the need for gasoline and jet fuels is increasing.

The consequences of these changes can be seen by comparing a refinery that processes high-quality, sweet (low-sulfur), light West Texas crude oil with one that processes a sour (high-sulfur), heavy Venezuelan crude oil. For the high-quality crude oils, the energy value of the products (jet fuel, gasoline, etc.) exiting the refinery is ~95% that of the crude oil entering the refinery. Some of these crude oils would operate, with some difficulty, in a car engine without refining. In contrast, for low-grade, heavy, more-plentiful (and cheaper) crude oils, the energy value of the products exiting the refinery is ~80% that of the crude oil entering the refinery. For coal liquefaction, the energy efficiency is ~60%. Much of the energy consumed within the refinery is used for converting lower-value hydrocarbon streams and natural gas into H_2. This H_2 is used for several purposes.

- *Production of light oil*. Heavy crude oils are removed from the ground at high temperatures and become highly viscous as they cool. Such oils are so viscous that they do not flow unless heated or dissolved in lighter oil. The very heavy oils have an H_2-to-carbon ratio as low as ~0.8. Refinery operations add H_2 to increase this ratio to between 1.5 and 2 (similar to gasoline), thereby yielding a light refinery oil product that can be separated into various transport fuels. The lighter the fuel (such as gasoline), the more H_2 that is required for this conversion. Light, high-quality crude oils have H_2-to-carbon ratios between 1.5 and 2. The transition from less-abundant light crude oils to more-abundant heavy crude oils implies a massive growth in H_2 demand.

- *Reduction of toxicity*. Oils contain a variety of carcinogenic compounds such as benzene (C_6H_6). Recent laws require that these substances be removed or destroyed. This can be accomplished by adding H_2 to convert these compounds to noncarcinogenic clean fuels.

- *Production of clean fuels*. A sour crude oil may be up to 6% sulfur by weight. Sulfur and other impurities are removed using H_2 to produce clean-burning fuels. Heavy oils tend to have much higher sulfur content. The sulfur is removed by using H_2 to convert it to hydrogen sulfide (H_2S), which is then oxidized to sulfur and sold as a by-product. Sulfur is removed to (1) avoid catalyst poisoning within the refinery, that interferes with refinery operations; (2) minimize corrosion in fuel transport and engines; (3) produce clean fuels; and (4) improve engine efficiency.

For the refineries, there are only two ways to change the H_2-to-carbon ratio of crude oil to make clean gasoline: remove carbon or add H_2. Carbon is removed by a process called coking, which involves the conversion of lower-value refinery streams into coke and H_2. The lower-value refinery streams are also converted to H_2 by steam reforming in which the carbon is released to the atmosphere as carbon dioxide. Alternatively, H_2 can be added to the crude oil from an outside source. This H_2 is usually produced from natural gas. In this case, the gasoline yield per barrel of crude oil increases because no crude oil components are used to make H_2. If an economic source of non-fossil H_2 could be developed, it would have three major impacts: (1) increased gasoline production per barrel of oil by avoiding use of some crude oil fractions to make H_2, (2) reduced imports of natural gas, and (3) avoidance of greenhouse gases from production of H_2.

Future Transportation Fuels and Hydrogen

Liquid fuels used for transportation account for 24% of the total world energy demand and 25% of the energy demand in the United States. In some high-technology states such as California, half of the total energy consumed is in the form of liquid fuels for transportation. The high fraction of the total energy used for transportation in California is a consequence of several factors (1) lifestyle, (1) an energy conservation ethic that has reduced heating and cooling loads, and (3) just-in-time manufacturing economy that is dependent on fast transportation systems.

Unless efficient, lightweight batteries or other energy storage devices are successfully developed, H_2 may be the route to use nuclear energy to meet transport fuel requirements. As discussed earlier, we have a growing H_2 economy that is hidden away at chemical plants and refineries. Meeting the refinery H_2 demand is a clear path forward for using nuclear power to meet some transport energy needs via this route. There is also the potential for H_2 to play a much larger role in transport fuels by further changes that may occur in liquid transport fuels.

- *Carbon-saver liquid fuels*. The concerns about the greenhouse effect have resulted in increased interest in liquid fuels that minimize carbon dioxide emissions per unit of energy delivered. This may be accomplished by further boosting the H_2-to-carbon ratio in the fuel to 2 or more. The production of such *carbon-saver* fuels from crude oils would require massive additional quantities of H_2. The H_2 would have to be made from non-fossil sources or the carbon dioxide from producing H_2 from natural gas would have to be sequestered. As a secondary effect, carbon-saver fuels would significantly increase gasoline, diesel, and jet fuel yields per barrel of oil.

- *Fuel cell liquid fuels*. Major research programs are under way to develop fuel cells as replacements for gasoline engines in automobiles. Hydrogen is the preferred fuel. Many of these proposed fuel-cell systems include an onboard system to convert a liquid fuel to H_2. This type of system takes advantage of the ease of transport and storage of liquid fuels while using more-efficient fuel cells. In such a future, a demand would exist for very clean liquid fuels with a high H_2 content to minimize onboard processing of the liquid fuel to H_2. Such fuels would be similar or perhaps identical to carbon-saver fuel. This would further accelerate the refinery demand for H_2.

  There are major implications in the use of any of these advanced fuels.

- *Feasibility*. These fuels could be used in existing engines. Changes are required at the refinery but not in the transport fleet. With existing refinery technology, large additions of H_2 to liquid fuels are possible today.

- *Strategic*. It may be feasible to produce fuels with 20% of the energy coming from the H_2 added at the refinery. If the H_2 is produced from domestic energy sources, this implies a 20% reduction in crude oil demand with resultant reductions in oil imports. If the H_2 is produced from nonfossil sources (or fossil sources with sequestration of carbon), it implies a similar reduction in carbon dioxide emissions. These options allow 20% of the transport fuel demand to be met by nuclear facilities producing H_2.

- *Hydrogen-economy transition*. It has been proposed to use H_2 directly as a transport fuel. There are three challenges: (1) good vehicle H_2 storage technologies, (2) economic methods to produce H_2, and (3) managing the transition between liquid fuels and H_2. Making a sudden transition from liquid fuels to H_2 would be an extraordinarily difficult task. However, carbon-saver and fuel-cells type fuels that are also compatible with existing engines, provide for such a transition. If 20% of the energy in transportation is indirectly from H_2, the manufacturing and pipeline infrastructure to produce that H_2 provides a credible approach for transition to an H_2-fuelled economy.

Chemical and Industrial Uses

As discussed earlier, H_2 is used to produce ammonia, various chemicals, and in other industrial processes. These H_2 applications will remain; however, there are several other potentially large applications. One of the largest potential H_2 industrial markets is as a replacement for carbon used in producing metals. Traditionally, iron ores (iron oxides) are converted to metallic iron using coke (carbon) as a reducing agent. In a few locations with low cost natural gas, the natural gas is used to make H_2 that directly reduces iron ores to iron metal. These processes are technologically very attractive because the iron does not contain the many impurities that were originally in the coal and ultimately remained with the iron. If carbon dioxide emissions must be limited or H_2 costs were low, H_2 would quickly replace coal for these large-scale applications.

Hydrogen Economics

Because most H_2 is currently made from natural gas, their costs are closely coupled. Large increases in natural gas prices have occurred in the last year. If these prices remain high, the chemical industries that consume large quantities of H_2 may move offshore to locations with lower-cost natural gas or seek alternative ways to produce H_2. A significant fraction of the refinery industry may also move offshore, particularly that part of the industry that uses imported—rather than domestic—crude oil. In the United States, the competition primarily comes from the Caribbean, where lower-cost natural gas is available.

The previous H_2 projections assume that United States natural gas prices will decrease to earlier levels. If this is not the case, the demand for H_2 to produce chemicals and fuels for the United States will grow; however, the facilities that use the H_2 may be built offshore. There are powerful balance-of-trade and economic incentives for the United States and other countries to find alternative methods of producing H_2.

If the technology can be developed, H_2 from nuclear facilities would be expected to become competitive with that from natural gas. Refineries and chemical plants have a nearly constant demand for H_2 that matches the base-load capabilities of nuclear facilities. The constant base-load demand for H_2 favors technologies with low fuel costs, such as nuclear energy. There are several other characteristics of the H_2 market that are well matched to nuclear energy.

- *Scale of operations*. The H_2 demand per customer is large. A 600-MW(t) reactor with 50% efficiency would produce ~75 million std ft^3/d—enough at present for a moderately large refinery processing an average crude oil. This quantity of H_2 is equal to only ~2% of current U.S. refinery H_2 production capacity. If carbon saver or fuel-cell fuels were to be produced, the H_2 demand per refinery would be many times larger.

- *Hydrogen transport*. In the United States, pipelines are located along the Gulf Coast for the transport of H_2 between refineries, chemical plants, and merchant-H_2 generation plants. Similar H_2 pipeline systems also exist in Europe. Hence, there is no requirement for collocation of refinery and H_2 production facilities. The pipelines also provide a means for industrial plants to import purchased H_2 when internal production facilities are down for maintenance, or to export H_2 for sale when internal consumption is low. Production economics, not the H_2 demand of a single consumer, would determine the H_2 plant size.

- *Finances*. Major oil companies each have annual sales in excess of a $100 billion per year. For industrial applications, the total cost of H_2 production (capital and operating) is the primary consideration. Front-end capital costs are not a barrier to the acceptance of a technology.

Nuclear Production Methods for H_2

Two approaches are being investigated for H_2 production using nuclear energy. The first approach is to use nuclear heat to assist in the production of H_2 from natural gas. Steam reforming of natural gas is an energy-intensive process where a fraction of the natural gas is used to provide heat at temperatures of up to 900°C. However, steam reforming of methane has carbon dioxide as a byproduct.

Alternatively, many direct thermochemical methods[4] are possible for producing H_2 with the input of heat and water. For low-production costs, however, high temperatures are required to ensure rapid chemical kinetics (small plant size with low capital costs) and high conversion efficiencies.[5,6] In addition, an attraction of several of the thermochemical processes is that there is no associated production of carbon dioxide.

Many types of thermochemical processes for H_2 production exist. The sulfuric acid processes (hydrogen sulfide, iodine–sulfur, and sulfuric acid–methanol) are the leading candidates. In each of these processes, the high-temperature, low-pressure endothermic (heat-absorbing) reaction is the thermal decomposition of sulfuric acid to produce oxygen:

$H_2SO_4 \Rightarrow H_2O + SO_2 + ½ O_2$ (high temperature).

Typically temperatures in the range of 800 to 1,000°C are needed for efficient H_2 production. After oxygen separation, additional chemical reactions are required to produce H_2. The leading candidate for thermochemical H_2 generation is the iodine–sulfur (IS) process, which has two additional chemical reactions:

$I_2 + SO_2 + 2H_2O \Rightarrow 2HI + H_2SO_4$ (low temperature)

and the H_2-producing step

$2HI \Rightarrow H_2 + I_2$ (intermediate temperature).

The Japan Atomic Energy Research Institute[6] is currently preparing to demonstrate the production of H_2 by steam reforming of natural gas with the heat-energy input provided by their High-Temperature Engineering Test Reactor (HTTR). The iodine–sulfur process is being developed with the ultimate goal to connect it to the HTTR. Research on this process is also under way in the United States. Significant development work on H_2 thermochemical cycles is required, with the technology being applicable to both nuclear and solar-power tower heat sources.

The economics of H_2 production strongly depends on the efficiency of the method used. Production efficiency can be defined as the energy content of the resulting H_2 divided by the energy expended to produce the H_2. Hydrogen production by electrolysis is relatively efficient (~80%). However, when this factor is combined with the electrical conversion efficiency, which ranges from approximately 34% (in current light-water reactors) to 50 % (for advanced systems), the overall efficiency would be approximately 25 to 40%. A significant capital investment in electrolytic cells is also required. For thermochemical approaches such as the iodine-sulfur process described previously, an overall efficiency of >50% has been projected. Combined-cycle (H_2 and electricity) plants may have efficiencies of ~60%. All of the efficient, potentially low-capital-cost thermochemical processes require high temperatures.

Reactor Requirements

Thermochemical production of H_2 imposes the following technical requirements on the reactor:

- *Temperature*. Temperatures of between 750 and 1,000°C are required. Higher temperatures are preferred.

- *Isolation*. Heat must be transferred from the nuclear system to the chemical plant at high temperatures. The intermediate heat transport system must be designed to decouple the nuclear reactor heat source and the thermochemical H_2 production plant such that an accident or upset in one part of the system will not propagate to the other. This requirement imposes still-to-be-defined constraints on the reactor.

Three reactor concepts have been identified that may be compatible with coupling to a thermochemical H_2 production facility.

- *High-temperature gas-cooled reactor (HTGR).* Many variants to the HTGR exist, including a pebble-bed reactor and a hexagonal fuel-block reactor.

- *Advanced high-temperature reactor (AHTR).* This is a modular molten-salt-cooled reactor that uses a coated-particle graphite-matrix fuel[7] The AHTR is similar to an HTGR except that high-pressure helium coolant is replaced with a low-pressure molten salt. Reactor exit coolant temperatures may be somewhat higher than those for the HTGR (better heat transfer with lower temperature differences between fuel and coolant), and the coolant operates at atmospheric pressure.

- *Lead-cooled fast reactor*. The operating temperatures are somewhat lower than those for the HTGR. Lead cooling is required because sodium boils at 883ºC—near the same temperatures that are required for H_2 production.

Summary

While much attention has been given to potential future uses of H_2, a robust H_2 economy does in fact already exists today and is linked to the worldwide chemical and refining industry. As a result, H_2 represents a potential existing market for nuclear energy that may ultimately grow to exceed that for electricity. If nuclear energy is to be a supplier of much of the world's future energy needs, the production of both electricity and H_2 need to be considered.

Appendix C

Other Concepts

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Other Concepts

As mentioned in the report, the four concepts that do not fall within strict Generation IV reactor definitions are described in this appendix.

TASSE—Thorium Based Accelerator Driven System

TASSE achieves the Generation IV goals by proposing an accelerator driven system with natural thorium mobile fuel.[8] The TASSE concept simplifies the fuel cycle and eliminates or reduces the fabrication and recycling. Enhancing the use of natural thorium resources, TASSE uses an external accelerator driven neutron source (ADS) to supplement the intrinsic neutron production, reduces the actinide stock provided from PWR and reducing strongly the radiotoxicity of wastes. The advantages of thorium fuel cycles have already been mentioned (Sections 3.1.1, 3.1.2, and 3.1.3). The concept is based on the use of mobile fuel with the ADS system allowing a once-through cycle or a critical system with recycling. Features that make TASSE interesting in considerations of Generation IV potential include:

- Use of mobile fuels (liquid metal, molten salt, or pebble bed)
- High temperature operation and high thermodynamic efficiency
- Inherent safety characteristics
- Fission product clean-up by online chemistry recycling or no recycling for the ADS option (once-through cycle with high burn-up and low radiotoxicity of waste for the ADS option)
- Flexibility to use various types of fuel composition.

TASSE reactor physics have been evaluated but technical issues of structural material corrosion, salt stability, and one line recycling remain to be investigated. Although uneconomical for electricity production, the ADS, in the TASSE concept, is viable as an alternative to recycling and enrichment of the fuel.

Electronuclear Reactors

Recently, revolutionary new nuclear reactions were claimed by Matsumoto, called Electro-Nuclear Reactions (ENRs). There were several kinds of ENRs and the most significant one was Electro-Nuclear Collapse (ENC). ENRs could be induced by simple techniques such as electrolysis or electric discharges in water. The two kinds of applications proposed are (1) building A new type of nuclear reactor (EN Reactor), which is based on ENRs; (2) using the ENR concept to effectively incinerate long-lived radioactive wastes.[9]

Converging Beam Neutron Source Reactors (CBNS)

The proposed concept here involves a modular, neutron-source driven subcritical fission.[10] The ultimate aim of the concept is to achieve a subcritical liquid-metal fast reactor operating at a k_{eff} of ~0.98 to 0.99. This reactor would be driven by multiple small-scale neutron sources arrayed throughout the core. The neutron sources are envisioned to be modular converging beam neutron sources (CBNSs), which are advanced versions of present day cylindrical Inertial-Electrostatic Confinement (IEC) devices that produce 14-MeV fast neutrons from a deuterium-tritium (D-T) fusion reaction. Neutron sources of this kind using converging beams of high-energy ions driven primarily by electrostatic forces have been in existence since the 1960s.

Accelerator Driven Hydrogen Production

Various types of ADS are being developed in many institutes. Most of them are based on the typical LMR core shape (pancake core) and designed to generate the electricity or to produce hydrogen. The ADS is believed to satisfy the goal of fuel utilization more than enough because it generates hydrogen by transmuting TRU from LWR spent fuel and makes the uranium in LWR spent fuel reusable. Excellent waste management potential, fuel usage flexibility, subcritical mode safety, and the fact that the accelerator drives the fission there is no need to separate actinides (except uranium) out leading to excellent weapons proliferation resistance. Accelerator reliability needs to be looked at if commercial interest is to be sought and although electricity production gives low payback here, the use of ADS nuclear systems may find a niche in hydrogen production for the future global economy.[11]

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